PART 2

SOUTHERN COMPANY

Southern Nuclear Operating Company
Vogtle Electric Generating Plant, Units 3 & 4
COL Application

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Vogtle Early Site Permit Application

Vogtle Electric Generating Plant, Units 3 & 4

COL Application

Part 2

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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

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CHAPTER 1
INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 INTRODUCTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

The Vogtle ESPA SSAR is hereby incorporated by reference into the COL application as described in Section 1.6.

Add the following paragraphs to the end of DCD Section 1.1

STD SUP 1.1-1 This Final Safety Analysis Report (FSAR) incorporates the Design Control Document (DCD) (as identified in Table 1.6-201) for a simplified passive advanced light water reactor plant provided by Westinghouse Electric Company, the entity originally sponsoring and obtaining the AP1000 design certification documented in 10 CFR Part 52, Appendix D. Throughout this FSAR, the "referenced DCD" is the AP1000 DCD submitted by Westinghouse as Revision 19 including any supplemental material as identified in Table 1.6-201. Unless otherwise specified, reference to the DCD refers to Tier 2 information, including references to the sensitive unclassified non-safeguards information (including proprietary information) and safeguards information, contained in the AP1000 DCD. Such DCD information is included in this combined license application in the same manner as it is included in the AP1000 DCD, i.e., references in the DCD are included as references in the FSAR, and material incorporated by reference into the DCD is incorporated by reference into the FSAR. Appropriate agreements are in place to provide for the licensee's rights to possession (including constructive possession) and use of the withheld sensitive unclassified non-safeguards information (including proprietary information) and safeguards information referenced in the AP1000 DCD for the life of the project.

Appendix D to 10 CFR Part 52 is hereby incorporated by reference into the COL application.

VEGP SUP 1.1-2 This Final Safety Analysis Report (FSAR) is hereby submitted under Section 103 of the Atomic Energy Act by the Southern Nuclear Operating Company, Inc. (SNC) to the NRC as part of the application for two Class 103 combined licenses (COLs) to construct and operate two nuclear power plants under the provisions of 10 CFR Part 52 Subpart C.

1.1.1 PLANT LOCATION

VEGP COL 2.1-1 Add the following text at the beginning of DCD Subsection 1.1.1:

VEGP Units 3 and 4 are located on a 3,169-acre coastal plain bluff on the southwest side of the Savannah River in eastern Burke County, Georgia. The site is approximately 30 river miles above the U.S. 301 bridge and directly across the river from the Department of Energy's Savannah River Site (Barnwell County, South Carolina). The VEGP site is approximately 15 miles east-northeast of Waynesboro, Georgia and 26 miles southeast of Augusta, Georgia. It is also about 100 miles from Savannah, Georgia and 150 river miles from the mouth of the Savannah River.

Figure 1.1-201 identifies the site location. Figure 1.1-202 shows the plant arrangement within the site.

1.1.5 SCHEDULE

Add the following text to the end of DCD Subsection 1.1.5:

VEGP COL 1.1-1 Table 1.1-203 displays the anticipated schedule for construction and operation of two AP1000 units at the VEGP site. A site-specific construction plan and startup schedule will be provided to the NRC after issuance of the COL.

1.1.6.1 Regulatory Guide 1.70

Add the following text to the end of DCD Subsection 1.1.6.1.

STD SUP 1.1-6 This FSAR generally follows the AP1000 DCD organization and numbering. Some organization and numbering differences are adopted where necessary to include additional material, such as additional content identified in Regulatory Guide 1.206. Any exceptions are identified with the appropriate left margin annotation as discussed in Subsection 1.1.6.3 and Table 1.1-202.

1.1.6.3 Text, Tables and Figures

Add the following text to the end of DCD Subsection 1.1.6.3.

STD SUP 1.1-3 Table 1.1-202 describes the left margin annotations used in this document to identify departures, supplementary information, COL items, and conceptual design information.

FSAR tables, figures, and references are numbered in the same manner as the DCD, but the first new FSAR item is numbered as 201, the second 202, the third 203, and consecutively thereafter. When a table, figure, or reference in the DCD is changed, the change is appropriately left margin annotated as identified above.

New appendices are included in the FSAR with double letter designations following the pertinent chapter (e.g., 12AA).

When it provides greater contextual clarity, an existing DCD table or figure is revised by adding new information to the table or figure and replacing the DCD table or figure with a new one in the FSAR. In this instance, the revised table or figure clearly identifies the information being added, and retains the same numbering as in the DCD, but the table or figure number is revised to end with the designation "R" to indicate that the table or figure has been revised and replaced. For example, revised "Table 4.2-1" would become "Table 4.2-1R." New and revised tables and figures are labeled in the left margin as described in Table 1.1-202.

1.1.6.5 Proprietary Information

Insert the following text to the end of DCD Subsection 1.1.6.5.

STD SUP 1.1-4 Some portions of this FSAR may be considered as proprietary, personal, or sensitive and withheld from public disclosure pursuant to 10 CFR 2.390 and Regulatory Issue Summary (RIS) 2005-026. Such material is clearly marked and the withheld material is separately provided for NRC review.

1.1.6.6 Acronyms

Add the following text to the end of DCD Subsection 1.1.6.6.

VEGP SUP 1.1-5 Table 1.1-201 provides a list of acronyms used in the VEGP Units 3 and 4 FSAR in addition to the acronyms identified in DCD Table 1.1-1 and system designation identified in Table 1.7-201 and DCD Table 1.7-2.

1.1.7 COMBINED LICENSE INFORMATION

Add the following after DCD Subsection 1.1.7.

VEGP COL 1.1-1 This COL item is addressed in Subsection 1.1.5.

VEGP SUP 1.1-5

Table 1.1-201 (Sheet 1 of 6)
Acronyms Used in the FSAR

Acronym	Definition
ACSR	Aluminum Conductor Steel Reinforced
AD	Air Diffuser
ADAMS	Agencywide Documents Access and Management System
AE	Architect Engineer
AFW	Auxiliary Feedwater
ALOHA	Areal Location of Hazardous Atmosphere
AR	Air Removal
ASS	Auxiliary Steam System
AWWA	American Water Works Association
BD	Bay Door
BDS	Blowdown System
BP	Blowout Panel
BPO	Bulk Power Operations
B&PVC	Boiler and Pressure Vessel Code
BR	Breathing Rate
BWR	Boiling Water Reactor
CAM	Continuous Air Monitor
CAPCO	Corrective Action Program Coordinator
CDI	Conceptual Design Information
CEO	Chief Executive Officer
CFO	Chief Financial Officer
CFS	Chemical Feed System
CN	Curve Number
CNO	Chief Nuclear Officer
COLA	Combined License Application
CP	Construction Phase
CR	Control Room
CS	Containment Shell
DAC	Derived Air Concentration
DAW	Dry Active Waste
DC	Direct Current

Table 1.1-201 (Sheet 2 of 6)
VEGP SUP 1.1-5
Acronyms Used in the FSAR

Acronym	Definition
DG	Diesel Generator
DRAP	Design Reliability Assurance Program
EAB	Exclusion Area Boundary
EAL	Emergency Action Level
EBR	Experimental Breeder Reactor
ECCS	Emergency Core Cooling System
ENS	Emergency Notification System
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EP	Emergency Planning
EPC	Engineering, Procurement & Construction
EP-ITAAC	Emergency Planning ITAAC
EQ	Environmental Qualification
EQMEL	EQ Master Equipment List
ERO	Emergency Response Organization
ESP	Early Site Permit
ESPA SSAR	Early Site Permit Application Site Safety Analysis Report
EVP	Executive Vice President
FAC	Flow Accelerated Corrosion
FERC	Federal Energy Regulatory Commission
FFD	Fitness For Duty
FHA	Fire Hazards Analysis
FMEA	Failure Mode Effects Analysis
FNP	Farley Nuclear Plant
FPS	Fire Protection System
FSAR	Final Safety Analysis Report
FSER	Final Safety Evaluation Report
FWS	Feedwater System
GCC	Georgia Transmission Control Center
GI-LLI	Gastrointestinal Tract–Lower Large Intestine
GPC	Georgia Power Company
GSU	Generator Step-up (Transformer)
HCLPF	High Confidence, Low Probability of Failure

Table 1.1-201 (Sheet 3 of 6)
VEGP SUP 1.1-5
Acronyms Used in the FSAR

Acronym	Definition
HNP	Hatch Nuclear Plant
HP	Health Physics
HV	High Voltage
IDLH	Immediately Dangerous to Life and Health
IIS	Incore Instrumentation System
INPO	Institute of Nuclear Plant Operations
ISFSI	Independent Spent Fuel Storage Installation
ITA	Inspections, Tests, Analyses
ITP	Initial Test Program
JOG	Joint Owners Group
JPM	Job Performance Measure
JTWG	Joint Test Working Group
LCO	Limiting Condition for Operation
LFL	Lower Flammability Limit
LLC	Limited Liability Corporation
LLW	Low Level Waste
LTOP	Low Temperature Overpressure Protection
MSL	Mean Sea Level
MSPI	Mitigating Systems Performance Indicator
NE	North East
NESC	National Electric Safety Code
ND	Nuclear Development
NDCT	Natural Draft Cooling Tower
NDE	Non-destructive Examination
NDQA	Nuclear Development Quality Assurance
NERC	North American Electric Reliability Corporation
NLO	Non-Licensed Operator
NPDES	National Pollutant Discharge Elimination System
NUS	Nuclear Utility Services
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
OE	Operating Experience
OJT	On-the-Job Training

Table 1.1-201 (Sheet 4 of 6)
VEGP SUP 1.1-5
Acronyms Used in the FSAR

Acronym	Definition
OM	Operations and Maintenance
OSC	Operations Support Center
PC	Permit Condition
PCC	Power Coordination Center
PCP	Process Control Program
PGS	Plant Gas System
PM	Project Manager
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PMWP	Probable Maximum Winter Precipitation
PNNL	Pacific Northwest National Laboratory
PORV	Power Operated Relief Valve
PR	Proposed Revision
PS-ITAAC	Physical Security ITAAC
PV	Plant Vent
PT&O	Plant Test and Operations
PVC	Polyvinyl Chloride
PWS	Potable Water System
PZR	Pressurizer
QAPD	Quality Assurance Program Description
QMS	Quality Management System
RAT	Reserve Auxiliary Transformer
RCA	Radiological Controlled Area
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RHR	Residual Heat Removal
RIS	Regulatory Issue Summary
RO	Reactor Operator
RP	Radiation Protection
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RS	Review Standard
RT	Radiography Techniques

Table 1.1-201 (Sheet 5 of 6)
VEGP SUP 1.1-5
Acronyms Used in the FSAR

Acronym	Definition
RTDP	Revised Thermal Design Procedure
SAMDA	Severe Accident Mitigation Design Alternatives
SAMG	Severe Accident Management Guidance
SAR	Safety Analysis Report
SAT	Systematic Approach to Training
SBAA	Southern Balancing Authority Area
SC	South Carolina
SCBA	Self Contained Breathing Apparatus
SCEG	South Carolina Electric and Gas
SCS	Southern Company Services
SCT	Southern Company Transmission
SDP	Significance Determination Process
SDS	Sanitary Drains System
SERC	South Eastern Reliability Corporation
SGMP	Steam Generator Management Program
SGS	Steam Generation System
SGTR	Steam Generator Tube Rupture
SGW/L	Steam Generator Water Leg
Shaw	Shaw Stone & Webster Nuclear Services
SNC	Southern Nuclear Operating Company
SNM	Special Nuclear Material
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SSC	Seismic Source Characterization
SSC(s)	Structure(s), System(s), and Component(s)
SS-ITAAC	Site-Specific ITAAC
STA	Shift Technical Advisor
SV	Steam Vent
SVP	Senior Vice President
TBD	To Be Determined
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent Dosimeter
TNT	Trinitrotoluene

Table 1.1-201 (Sheet 6 of 6)
VEGP SUP 1.1-5
Acronyms Used in the FSAR

Acronym	Definition
TS	Technical Specification(s)
TSO	Transmission System Operator
TSP	Transmission System Provider
TtNUS	Tetra Tech NUS, Inc.
UAT	Unit Auxiliary Transformer
UFL	Upper Flammability Limit
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Techniques
V & V	Verification and Validation
VEGP	Vogtle Electric Generating Plant
VP	Vice President
WAC	Waste Acceptance Criteria
WEC	Westinghouse Electric Company
WMA	Wildlife Management Area

STD SUP 1.1-3

Table 1.1-202 (Sheet 1 of 2)
Left Margin Annotations

MARGIN NOTATION	DEFINITION AND USE
STD DEP X.Y.Z-#	FSAR information that departs from the generic DCD and is common for parallel applicants. Each Standard Departure is numbered separately at an appropriate level, e.g., STD DEP 9.2-1, or STD DEP 9.2.1-1
NPP DEP X.Y.Z-#	FSAR information that departs from the generic DCD and is plant specific. NPP is replaced with a plant specific identifier. Each Departure item is numbered separately at an appropriate subsection level, e.g., NPP DEP 9.2-2, or NPP DEP 9.2.1-2
STD COL X.Y-#	FSAR information that addresses a DCD Combined License Information item and is common to other COL applicants. Each COL item is numbered as identified in DCD Table 1.8-2 and FSAR Table 1.8-202, e.g., STD COL 4.4-1, or STD COL 19.59.10.5-1
NPP COL X.Y-#	FSAR information that addresses a DCD Combined License Information item and is plant specific. NPP is replaced with a plant specific identifier. Each COL item is numbered as identified in DCD Table 1.8-2 and FSAR Table 1.8-202, e.g., NPP COL 4.4-1, or NPP COL 19.59.10.5-1
NPP CDI or STD CDI	FSAR information that addresses DCD Conceptual Design Information (CDI). Replacement design information is generally plant specific; however, some may be common to other applicants. NPP is replaced with a plant specific identifier. STD is used if it is common. CDI information replacements are not numbered.

STD SUP 1.1-3

Table 1.1-202 (Sheet 2 of 2)
Left Margin Annotations

STD SUP X.Y-#	FSAR information that supplements the material in the DCD and is common to other COL applicants. Each SUP item is numbered separately at an appropriate subsection level, e.g., STD SUP 1.10-1, or STD SUP 9.5.1-1
NPP SUP X.Y-#	FSAR information that supplements the material in the DCD and is plant specific. NPP is replaced with a plant specific identifier. Each SUP item is numbered separately at an appropriate subsection level, e.g., NPP SUP 3.10-1, or NPP SUP 9.2.5-1
DCD	FSAR information that duplicates material in the DCD. Such information from the DCD is repeated in the FSAR only in instances determined necessary to provide contextual clarity.
NPP ESP PC#	FSAR information that addresses an ESP Permit Condition. NPP is replaced with a plant specific identifier. An ESP Permit Condition is numbered as identified in the applicable ESP.
NPP ESP VAR X.Y-#	A request for an ESP Variance. NPP is replaced with a plant specific identifier. Each ESP Variance is numbered based on the applicable section down to the ESP Application X.Y.Z level.
NPP ESP COL X.Y-#	FSAR information that addresses an ESP COL action item. NPP is replaced with a plant specific identifier. An ESP COL information item is numbered as identified in the applicable ESP down to the X.Y level.

VEGP SUP 1.1-8

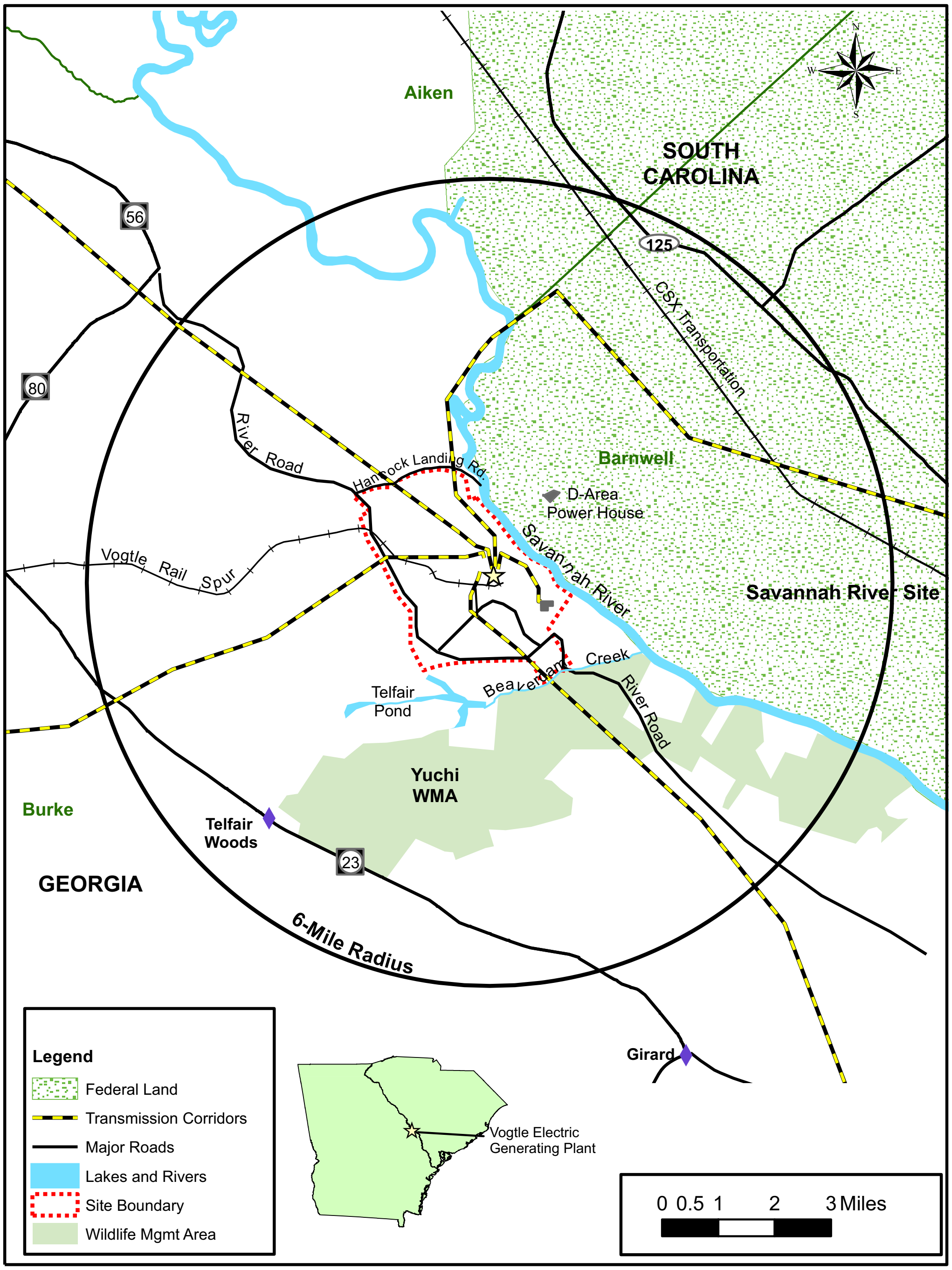
VEGP COL 1.1-1

Table 1.1-203
Anticipated Schedule for Construction and Operation of Two AP1000 Units at the VEGP Site

Activity	Start(1)	Finish(1)	Duration
UNIT 3			
Early Procurement Activities	2nd Q 2008	--	--
Site Preparation	3rd Q 2008	3rd Q 2011	36 Months
Commence Construction (Safety-related activities)	1st Q 2010 (LWA)		--
Fuel Load, Commence Start-Up	4th Q 2015	2nd Q 2016	6 Months
Commercial Operation	April 2016		
UNIT 4			
Early Procurement Activities	2nd Q 2008	--	--
Site Preparation	3rd Q 2008	3rd Q 2012	48 Months
Commence Construction (Safety-related activities)	1st Q 2010 (LWA)		--
Fuel Load, Commence Start-Up	4th Q 2016	2nd Q 2017	6 Months
Commercial Operation	April 2017		

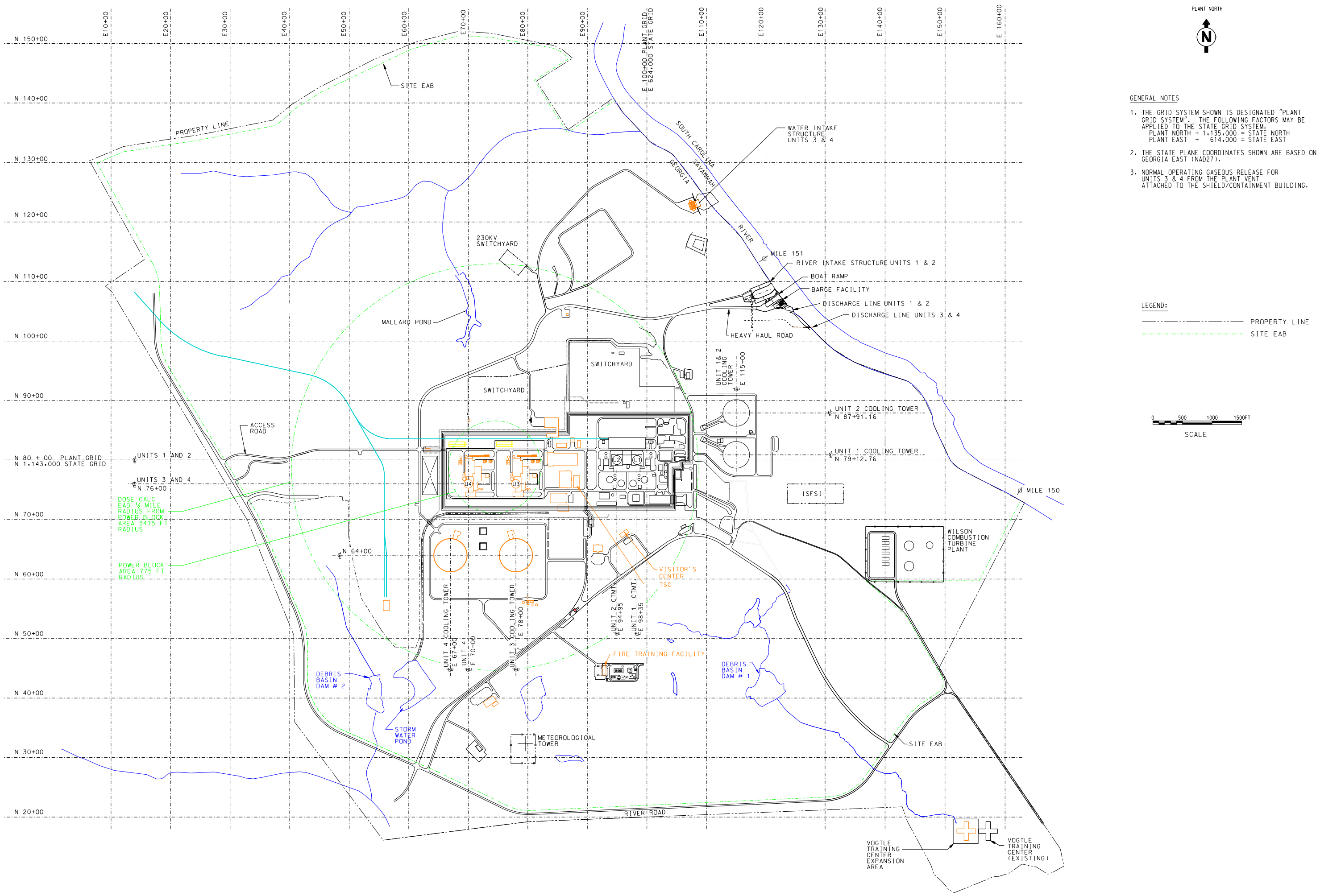
Notes:

(1) Activities that do not indicate a start or finish date are milestones, and represent the anticipated commencement (start) or completion (finish) of the activity.



VEGP COL 2.1-1

Figure 1.1-201
Site Location Map



VEGP ESP VAR 1.2-1
VEGP COL 2.1-1
VEGP DEP 18.8-1

Figure 1.1-202
Site Layout

1.2 GENERAL PLANT DESCRIPTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

This section of the referenced ESPA SSAR is incorporated by reference with no variances or supplements.

1.2.2 SITE DESCRIPTION

In Subsection 1.2.2 of the DCD, replace the information entitled “Site Plan” with the following text.

VEGP COL 2.1-1
VEGP COL 3.3-1
VEGP COL 3.5-1

Site Plan

The VEGP site plan is described in the referenced ESPA SSAR.

The site layout is shown in Figure 1.1-202. Principal structures, facilities, and roads are illustrated. Orientation of the two AP1000 units is such that “plant north” corresponds with true north. Plant elevation in the DCD is 100 feet, whereas the plant elevation for VEGP is 220 feet; therefore DCD elevations are to be increased by 120 feet to be actual site elevations.

The river intake for VEGP Units 3 and 4 withdraw makeup water from the Savannah River. The intake system is located upstream of the river intake of VEGP Units 1 and 2. Makeup water is pumped directly to the cooling tower basin. Cooling tower circulating water pump complexes are located south of the reactors. The intake structure is located at the end of a canal, and is situated northeast of the reactors.

The transformer area is located immediately adjacent to and north of the turbine building. The unit auxiliary transformers (UAT), the reserve auxiliary transformers (RAT), and the generator step-up transformers (GSU) are located in this area. The main switchyard area is located adjacent to the transformer area.

An administrative building is located east of the cooling tower basins. The building houses management, administration, and support functions for the site. The training and simulator building is located southeast of the reactor complex, but within the site boundary.

1.2.3 PLANT ARRANGEMENT DESCRIPTION

Add the following information at the end of the first paragraph of DCD Subsection 1.2.3.

VEGP DEP 18.8-1 Figure 1.2-201 replaces DCD Figure 1.2-18 to reflect the relocation of the Operations Support Center.

Security-Related Information — Withheld Under 10 CFR 2.390(d)
(See Part 9 of this COL Application)

(Note: This figure replaces DCD Figure 1.2-18.)

VEGP DEP 18.8-1

Figure 1.2-201
Annex Building General Arrangement
Plan at Elevation 100'-0" & 107'-2"

1.3 COMPARISONS WITH SIMILAR FACILITY DESIGNS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

This section of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

1.4.1 APPLICANT – PROGRAM MANAGER

Add the following paragraphs as the first three paragraphs in DCD Subsection 1.4.1:

VEGP SUP 1.4-1 The Southern Nuclear Operating Company, Inc. (SNC) is the non-owner applicant for a Combined License for the VEGP and will be the constructor and licensed operator of Units 3 and 4. The owner licensees are as follows: Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, Georgia, an incorporated municipality in the State of Georgia acting by and through its Board of Water, Light and Sinking Fund Commissioners (Dalton Utilities). SNC is a wholly-owned subsidiary of Southern Company.

SNC was formed for the purpose of operating nuclear facilities owned by other Southern Company subsidiaries. SNC operates the Edwin I. Hatch Nuclear Plant, Units 1 and 2, the Vogtle Electric Generating Plant, Units 1 and 2, and the Joseph M. Farley Nuclear Plant Units 1 and 2. The combined electric generation of the three plants is in excess of 5,900 MWe.

On April 8, 2008, the owner licensees for VEGP Units 3 and 4 executed a contract for Engineering, Procurement, and Construction (EPC) of the facilities with a Consortium comprised of Westinghouse and Stone & Webster, Inc. (also referred to herein as Shaw Stone & Webster or simply Shaw). The Consortium will act as the AP1000 provider and architect-engineer for VEGP Units 3 and 4. Southern Nuclear, as the constructor of VEGP Units 3 and 4, has delegated responsibility for physical construction activities to the Consortium.

Add the following paragraphs to the end of DCD Subsection 1.4.1:

VEGP SUP 1.4-2 Shaw is a Fortune 500 company which has been an active participant in the nuclear industry for nearly 60 years, from providing engineering and design services for Shippingsport, the nation's first commercial nuclear power plant, to the restart of Tennessee Valley Authority's Browns Ferry Unit 1, which at the time was the largest nuclear construction project in the western hemisphere. Shaw continues to prove its leadership role in the nuclear industry by being part of the

AP1000 Consortium. Shaw is part of a vertically integrated company, Shaw Group, Inc., which has nearly 180 offices worldwide and over 21,000 employees, of which approximately 3,100 are nuclear professionals offering nuclear services on four continents.

Westinghouse is responsible for the overall plant design, AP1000 Design Certification revisions, procurement of primary NSSS equipment and power block major components including the Turbine Generator, and plant training simulator. Shaw is responsible for site development, construction, site specific design related work, secondary equipment procurement, module fabrication, and supply of bulk materials and commodities. Westinghouse and Shaw are jointly responsible for testing and startup. Fuel supply will be provided by Westinghouse under a separate contract.

Add the following new subsection after DCD Subsection 1.4.2.7:

VEGP SUP 1.4-3 1.4.2.8 Other Contractors

Add ESPA SSAR Section 1.4 to Subsection 1.4.2.8 with the following new first sentence prior to the first paragraph:

The following contractors also assisted in the preparation of the COL application.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

This section of the referenced ESPA SSAR is incorporated by reference with no variances or supplements.

1.6 MATERIAL REFERENCED

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following text to the end of DCD Section 1.6.

STD SUP 1.6-1 Table 1.6-201 provides a list of the various technical documents incorporated by reference in the FSAR in addition to those technical documents incorporated by reference in the AP1000 DCD.

VEGP SUP 1.6-2 The Vogtle Early Site Permit Application (ESPA) Site Safety Analysis Report (SSAR) is incorporated by reference into this Combined License Application (COLA) Final Safety Analysis Report (FSAR) with variances and/or supplements as noted. Table 1.6-202, Cross Reference of SSAR Sections Incorporated by Reference into FSAR Sections, provides information regarding incorporation of SSAR information into the FSAR.

Reference to the ESPA SSAR is understood to mean Revision 5, as submitted by Southern Nuclear Operating Company (SNC) on December 23, 2008, and as approved by the NRC in the Vogtle Early Site Permit and Limited Work Authorization (ESP-004), dated August 26, 2009 (ADAMS Accession Numbers ML092290130 and ML092290157) including the following three Amendments as identified below:

- Amendment 1 to Early Site Permit No. ESP-004, dated May 21, 2010 (ADAMS Accession Number ML101400509)
- Amendment 2 to Early Site Permit No. ESP-004, dated June 25, 2010 (ADAMS Accession Number ML101760370)
- Amendment 3 to Early Site Permit No. ESP-004, dated July 9, 2010 (ADAMS Accession Number ML101870522)

Table 1.6-201
STD SUP 1.6-1 Additional Material Referenced

	Author/ Report Number[a]	Title	Revision	FSAR Section	Document Transmittal	ADAMS Accession Number
VEGP SUP 1.6-2	VEGP ESPA SSAR	Vogtle Early Site Permit Application Site Safety Analysis Report as approved by the NRC in the Vogtle Early Site Permit and Limited Work Authorization (ESP-004), dated August 26, 2009, including Amendments 1, 2 and 3.	5	Table 1.6-202	December 2008	ML090280033
	Westinghouse/ APP-GW-GL-700	AP1000 Design Control Document	19	All	June 2011	ML11171A500
	NEI 07-08A	Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)	0	12.1	October 2009	ML093220164
	NEI 07-03A	Generic FSAR Template Guidance for Radiation Protection Program Description	0	Appendix 12AA	May 2009	ML091490684
	NEI 06-13A	Template for an Industry Training Program Description	2	13.2	March 2009	ML090910554
	NEI 07-02A	Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52	0	17.6	March 2008	ML080910149
	10 CFR Part 52 Appendix D	Design Certification Rule for the AP1000 Design	–	1.1	–	–
VEGP SUP 1.6-3	QAPD	SNC Nuclear Development Quality Assurance Manual	9.0	17.5	December 2009	ML100081148
	Emergency Plan	VEGP 3 and 4 Emergency Plan	4	13.3	January 2011	ML110390284
	Security Plan	VEGP 3 and 4 Physical Security Plan	2	13.6	July 2010	Not Applicable (Safeguards)
	Cyber Security Plan	VEGP 3 and 4 Cyber Security Plan	1	13.6	June 2011	Not Applicable (SUNSI)

a) The NRC-accepted NEI documents identified by the A in the document number include the accepted template, the NRC safety evaluation, and corresponding responses to the NRC Requests for Additional Information. Only the accepted template is incorporated by reference. The remainder of the document is referenced but not incorporated into the FSAR.

(A) Denotes NRC approved document.

Table 1.6-202 (Sheet 1 of 4)
Cross Reference of ESPA SSAR Sections Incorporated by Reference into FSAR Sections

VEGP SUP 1.6-2
(Unless Otherwise Noted)

	SSAR Section	SSAR Section Title	Corresponding FSAR Section
	1.1	Introduction	SSAR Section 1.1 provides general information related to the ESP proceeding, and is not applicable to any particular FSAR section.
VEGP ESP VAR 1.2-1	1.2	General Site Description	Section 1.1.1 Plant Location. This ESPA SSAR Section is Incorporated by Reference into FSAR Subsection 1.1.1 with the exception of Figures 1-4 and 1-5. COLA Part 7 requests a variance for this ESPA section.
VEGP ESP VAR 2.3-1	1.3	Site Characteristics, Design Parameters, and Site Interface Values	Section 2.0, Site Characteristics. This ESPA SSAR Section is Incorporated by Reference into FSAR Section 2.0 with the exception of Table 1-1 values for Maximum Normal Dry- and Wet-Bulb temperatures and Minimum Dry Bulb temperature. COLA Part 7 requests a variance for this ESPA table.
	1.4	Identification of Agents and Contractors	Section 1.4, Identification of Agents and Contractors
	1.5	Requirements for Further Technical Information	Section 1.5, Requirements for Further Technical Information
VEGP ESP VAR 1.6-1	1.6	Material Incorporated by Reference	This ESPA SSAR section is not Incorporated by Reference into the FSAR. This section of the ESPA SSAR includes a reference to Revision 15 of the AP1000 DCD. COLA Part 7 requests a variance for this ESPA section.
	1.7	Drawings and Other Detailed Information	Section 1.7, Drawings and Other Detailed Information

Table 1.6-202 (Sheet 2 of 4)
Cross Reference of ESPA SSAR Sections Incorporated by Reference into FSAR Sections

VEGP SUP 1.6-2
(Unless Otherwise Noted)

	SSAR Section	SSAR Section Title	Corresponding FSAR Section
	1.8	Conformance to NRC Regulations and Regulatory Guidance	Section 1.9, Compliance With Regulatory Criteria.
	2.1	Geography and Demography	Section 2.1, Geography and Demography
VEGP ESP VAR 2.2-1	2.2	Identification of Potential Hazards in Site Vicinity	Section 2.2, Nearby Industrial, Transportation, and Military Facilities. This ESPA SSAR Section is Incorporated by Reference into FSAR Section 2.2 with the exception of the last paragraph of ESPA SSAR 2.2.3.2.3, and ESPA SSAR Table 2.2-6. This information has been superseded by information contained in Sections 2.2 and 6.4. COLA Part 7 requests a variance for this ESPA subsection and table.
VEGP ESP VAR 2.3-1	2.3	Meteorology	Section 2.3, Meteorology. The third from last and second from last paragraphs of ESPA SSAR Subsection 2.3.1.5 are replaced by information contained in the replacement paragraph, which is shown in Subsection 2.3.1.5. COLA Part 7 requests a variance for this ESPA subsection.
	2.4	Hydrologic Engineering	Section 2.4, Hydrologic Engineering
	2.5	Geology, Seismology, and Geotechnical Engineering	Section 2.5, Geology, Seismology, and Geotechnical Information
	3.5.1.6	Aircraft Hazards	Section 3.5.1.6, Aircraft Hazards

Table 1.6-202 (Sheet 3 of 4)
Cross Reference of ESPA SSAR Sections Incorporated by Reference into FSAR Sections

VEGP SUP 1.6-2
(Unless Otherwise Noted)

	SSAR Section	SSAR Section Title	Corresponding FSAR Section
VEGP ESP VAR 1.6-2	3.8.5	Foundations	This ESPA SSAR subsection is Incorporated by Reference into FSAR Subsection 3.8.5.1 with the exception of the first paragraph. This paragraph includes a reference to Revision 15 of the AP1000 DCD. Additionally, the first paragraph in ESPA SSAR Subsection 3.8.5.1.1 is not incorporated by reference. COLA Part 7 requests a variance for this ESPA section.
	11.2.3	Liquid Radioactive Releases	Section 11.2.3.5, Estimated Doses
	11.3.3	Gaseous Radioactive Releases	Section 11.3.3.4, Estimated Doses
VEGP ESP VAR 1.2-1	13.3	Emergency Planning	Section 13.3, Emergency Planning. This ESPA SSAR section is Incorporated by Reference into FSAR Section 13.3 with the exception of Figure 13.3-2. COLA Part 7 requests a variance for this ESPA section.
	13.6	Industrial Security	Section 13.6, Security
	13.7	Fitness for Duty	Section 13.7, Fitness for Duty
VEGP ESP VAR 1.6-3	15	Accident Analyses	This ESPA SSAR chapter is not Incorporated by Reference into the FSAR. This chapter of the ESPA SSAR provides accident release information that has been superseded by the referenced DCD. COLA Part 7 requests a variance for this ESPA section.

Table 1.6-202 (Sheet 4 of 4)
Cross Reference of ESPA SSAR Sections Incorporated by Reference into FSAR Sections

VEGP SUP 1.6-2
(Unless Otherwise Noted)

SSAR Section	SSAR Section Title	Corresponding FSAR Section
17	Quality Assurance	This ESPA SSAR chapter is not Incorporated by Reference into the FSAR. The SNC Nuclear Development Quality Assurance Manual, provided as Appendix 17.1A of the ESPA SSAR, is now provided in COLA Part 11.

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

This section of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

1.7.2 PIPING AND INSTRUMENTATION DIAGRAMS

Add the following text to the end of DCD Subsection 1.7.2.

VEGP SUP 1.7-1 Table 1.7-201 contains a list of piping and instrumentation diagrams (P&IDs) or system diagrams and the corresponding FSAR figure numbers that supplement the DCD.

VEGP SUP 1.7-1

Table 1.7-201
AP1000 System Designators and System Drawings

Designator	System	FSAR Section	FSAR Figure
CWS	Circulating Water System	10.4.5	10.4-201
RWS	Raw Water System	9.2.11	9.2-201
ZBS	Offsite Power System One-Line Diagram	8.2	8.2-201
ZBS	Switchyard General Arrangement	8.2	8.2-202

1.8 INTERFACES FOR STANDARD DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following paragraphs to the end of DCD Section 1.8.

VEGP SUP 1.8-1 Departures from the referenced DCD are summarized in Table 1.8-201. Table 1.8-201 lists each departure and the FSAR section or subsection impacted.

Variances from the referenced ESPA are idenified in Table 1.6-202.

VEGP SUP 1.8-2 DCD Table 1.8-2 presents Combined License Information for the AP1000. Items requiring COL Applicant or COL Holder action are presented in Table 1.8-202. FSAR section(s) addressing these COL items are tabulated in this table. COL Holder items listed in Table 1.8-202 are regulatory commitments of the COL Holder and these actions will be completed as specified in the appropriate section of the referenced DCD. Completion of these COL Holder items is the subject of a proposed License Condition as presented in a separate document submitted as part of this COL application.

VEGP SUP 1.8-3 Table 1.8-203 lists the ESP COL action items and the corresponding FSAR section(s) that address these COL action items. ESP COL action items that are not addressed in the FSAR are not identified on Table 1.8-203. These ESP COL action items will be resolved through the ESP proceedings.

VEGP SUP 1.8-5 Table 1.8-204 lists the ESP permit conditions and the corresponding locations that address these permit conditions.

VEGP SUP 1.8-4 Demonstrations that the VEGP Units 3 and 4 site characteristics, design parameters, and site interface values fall within the site-related parameters for which the AP1000 was designed are provided in FSAR Section 2.0.

VEGP SUP 1.8-6 DCD Table 1.8-1 presents interface items for the AP1000. FSAR section(s) addressing these interface items are tabulated in Table 1.8-205.

VEGP SUP 1.8-1

Table 1.8-201 (Sheet 1 of 2)
Summary of FSAR Departures from the DCD

Departure Number	Departure Description Summary	FSAR Section or Subsection
VEGP DEP 1.1-1	An administrative departure is established to identify instances where the renumbering of FSAR sections is necessary to effectively include content consistent with Regulatory Guide 1.206, as well as NUREG-0800.	2.1.1 2.1.4 2.2.1 2.2.4 2.4.1 2.4.15 2.5 2.5.7 9.2.11 9.2.12 9.2.13 9.5.1.8 9.5.1.9 13.1 13.1.4 13.3.6 13.5 13.5.3 13.7 17.5 17.6 17.7 17.8
VEGP DEP 2.5-1	The lower and upper mudmat thickness is as presented in the ESPA SSAR	2.5.4.1.3
VEGP DEP 3.4-1	An alternate waterproofing system for the seismic Category I structures below grade is as presented in the ESPA SSAR.	3.4.1.1.1.1
STD DEP 8.3-1	The Class 1E voltage regulating transformers do not have active components to limit current.	8.3.2.2
VEGP DEP 9.2-1	The water source to the Potable Water System does not require filtration.	9.2.5.3

Table 1.8-201 (Sheet 2 of 2)
VEGP SUP 1.8-1 Summary of FSAR Departures from the DCD

Departure Number	Departure Description Summary	FSAR Section or Subsection
VEGP DEP 18.8-1	At VEGP, the Technical Support Center (TSC) is not located in the control support area (CSA) as identified in DCD Subsection 18.8.3.5; the TSC location is as described in the Emergency Plan. Additionally, the Operations Support Center (OSC) is also being moved from the location identified in DCD Subsections 18.8.3.6 and 12.5.2.2 and as identified on DCD figures in Subsections 1.2 and 12.3, and Appendix 9A; the OSC location is as described in the Emergency Plan.	1.2.3 9A 12.3 12.5.2.2 13.3.8 18.8.3.5 18.8.3.6

Table 1.8-202 (Sheet 1 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
1.1-1	Construction and Startup Schedule	1.1.7	1.1.5 1.1.7	A
1.9-1	Regulatory Guide Conformance	1.9.1.5	1.9.1 1.9.1.1 1.9.1.2 1.9.1.3 1.9.1.4 1.9.1.5 Appendix 1A Appendix 1AA	A
1.9-2[a]	Bulletins and Generic Letters	1.9.5.5	1.9.5.5	A
1.9-3[a]	Unresolved Safety Issues and Generic Safety Issues	Table 1.9-2 1.9.4.1	1.9.4.1 1.9.4.2.3	A
2.1-1	Geography and Demography	2.1.1	1.1.1 1.2.2 2.1.4	A
2.2-1	Identification of Site-specific Potential Hazards	2.2.1	2.2.3.2.3.1 2.2.3.2.3.2 2.2.3.3 2.2.3.4 2.2.4	A
2.3-1	Regional Climatology	2.3.6.1	2.3.6.1	A

Table 1.8-202 (Sheet 2 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
2.3-2	Local Meteorology	2.3.6.2	2.3.6.2	A
2.3-3	Onsite Meteorological Measurements Program	2.3.6.3	2.3.3.4 2.3.6.3	A
2.3-4	Short-Term Diffusion Estimates	2.3.6.4	2.3.4 2.3.6.4 15.6.5.3.7.3 15A.3.3	A
2.3-5	Long-Term Diffusion Estimates	2.3.6.5	2.3.5 2.3.6.5	A
2.4-1	Hydrological Description	2.4.1.1	2.4.15.1	A
2.4-2	Floods	2.4.1.2	2.4.2 2.4.10 2.4.15.2	A
2.4-3	Cooling Water Supply	2.4.1.3	2.4.15.3	A
2.4-4	Groundwater	2.4.1.4	2.4.15.4	A
2.4-5	Accidental Release of Liquid Effluents into Ground and Surface Water	2.4.1.5	2.4.15.5	A
2.4-6	Flood Protection Emergency Operation Procedures	2.4.1.6	2.4.14 2.4.15.6	A

Table 1.8-202 (Sheet 3 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
2.5-1	Basic Geologic and Seismic Information	2.5.1	2.5.7.1	A
2.5-2	Site Seismic and Tectonic Characteristics Information	2.5.2.1	2.5.7.2	A
2.5-3	Geoscience Parameters	2.5.2.3	2.5.7.3	A
2.5-4	Surface Faulting	2.5.3	2.5.7.4	A
2.5-5	Site and Structures	2.5.4.6.1	2.5.7.5	A
2.5-6	Properties of Underlying Materials	2.5.4.6.2	2.5.7.6	A
2.5-7	Excavation and Backfill	2.5.4.6.3	2.5.7.7	A
2.5-8	Ground Water Conditions	2.5.4.6.4	2.5.7.8	A
2.5-9	Liquefaction Potential	2.5.4.6.5	2.5.7.9	A
2.5-10	Bearing Capacity	2.5.4.6.6	2.5.7.10	A
2.5-11	Earth Pressures	2.5.4.6.7	2.5.7.11	A
2.5-12	Static and Dynamic Stability of Facilities	2.5.4.6.9	2.5.7.12	A
2.5-13	Subsurface Instrumentation	2.5.4.6.10	2.5.7.13	A
2.5-14	Stability of Slopes	2.5.5	2.5.7.14	A
2.5-15	Embankments and Dams	2.5.6	2.5.7.15	A

Table 1.8-202 (Sheet 4 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
2.5-16	Settlement of Nuclear Island	2.5.4.6.11	2.5.7.16	A
2.5-17	Waterproofing System	2.5.4.6.12	2.5.7.17 3.4.1.1.1.1 3.8.5.1	A
3.3-1	Wind and Tornado Site Interface Criteria	3.3.3	1.2.2 3.3.1.1 3.3.2.1 3.3.2.3 3.3.3 3.5.1.5 3.5.1.6	A
3.4-1	Site-Specific Flooding Hazards Protective Measures	3.4.3	3.4.1.3 3.4.3	A
3.5-1	External Missile Protection Requirements	3.5.4	1.2.2 3.3.1.1 3.3.2.1 3.3.2.3 3.5.1.5 3.5.1.6 3.5.4	A
3.6-1	Pipe Break Hazards Analysis	3.6.4.1	3.6.4.1 14.3.3.2	H

Table 1.8-202 (Sheet 5 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
3.6-4	Primary System Inspection Program for Leak-Before-Break Piping	3.6.4.4	3.6.4.4	A
3.7-1	Seismic Analysis of Dams	3.7.5.1	3.7.2.12 3.7.5.1	A
3.7-2	Post-Earthquake Procedures	3.7.5.2	3.7.4.4 3.7.5.2	A
3.7-3	Seismic Interaction Review	3.7.5.3	3.7.5.3	H
3.7-4	Reconciliation of Seismic Analyses of Nuclear Island Structures	3.7.5.4	3.7.5.4	H
3.7-5	Location of Free-Field Acceleration Sensor	3.7.5.5	3.7.4.2.1 3.7.5.5	A
3.8-5	Structures Inspection Program	3.8.6.5	3.8.3.7 3.8.4.7 3.8.5.7 3.8.6.5 17.6	A
3.8-6	Construction Procedures Program	3.8.6.6	3.8.6.6	H
3.9-2	Design Specification and Reports	3.9.8.2	3.9.8.2	H
3.9-3	Snubber Operability Testing	3.9.8.3	3.9.3.4.4 3.9.8.3	A

Table 1.8-202 (Sheet 6 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
3.9-4	Valve Inservice Testing	3.9.8.4	3.9.6 3.9.6.2.2 3.9.6.2.4 3.9.6.2.5 3.9.6.3 3.9.8.4	A
3.9-5	Surge Line Thermal Monitoring	3.9.8.5	3.9.3.1.2 3.9.8.5 14.2.9.2.22	A
3.9-7	As-Designed Piping Analysis	3.9.8.7	3.9.8.7 14.3.3.3	H
3.11-1	Equipment Qualification File	3.11.5	3.11.5	H
4.4-2	Confirm Assumptions for Safety Analyses DNBR Limits	4.4.7.2	4.4.7	H
5.2-1	ASME Code and Addenda	5.2.6.1	5.2.1.1 5.2.6.1	A

Table 1.8-202 (Sheet 7 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
5.2-2	Plant Specific Inspection Program	5.2.6.2	5.2.4 5.2.4.1 5.2.4.3.1 5.2.4.3.2 5.2.4.4 5.2.4.5 5.2.4.6 5.2.4.8 5.2.4.9 5.2.4.10 5.2.6.2	A
5.2-3	Response to Unidentified Reactor Coolant System Leakage Inside Containment	5.2.6.3	5.2.6.3 5.2.5.3.5	A
5.3-1	Reactor Vessel Pressure – Temperature Limit Curves	5.3.6.1	5.3.6.1	H
5.3-2	Reactor Vessel Materials Surveillance Program	5.3.6.2	5.3.2.6 5.3.2.6.3 5.3.6.2	A
5.3-4	Reactor Vessel Materials Properties Verification	5.3.6.4.1	5.3.6.4.1	H
5.3-7	Quickloc Weld Build-up ISI	5.3.6.6	5.2.4.1 5.3.6.6	A
5.4-1	Steam Generator Tube Integrity	5.4.15	5.4.2.5 5.4.15	A

Table 1.8-202 (Sheet 8 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
6.1-1	Procedure Review for Austenitic Stainless Steels	6.1.3.1	6.1.1.2 6.1.3.1	A
6.1-2	Coating Program	6.1.3.2	6.1.2.1.6 6.1.3.2	A
6.2-1	Containment Leak Rate Testing	6.2.6	6.2.5.1 6.2.5.2.2 6.2.6	A
6.3-1	Containment Cleanliness Program	6.3.8.1	6.3.8.1	A
6.4-1	Local Hazardous Gas Services and Monitoring	6.4.7	2.2.3.2.3.1 2.2.3.2.3.2 2.2.3.3 6.4.4.2 6.4.7	A
6.4-2	Procedures for Training for Control Room Habitability	6.4.7	6.4.3 6.4.7	A
6.6-1	Inspection Programs	6.6.9.1	6.6 6.6.1 6.6.3.1 6.6.3.2 6.6.3.3 6.6.4 6.6.6 6.6.9.1	A

Table 1.8-202 (Sheet 9 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
6.6-2	Construction Activities	6.6.9.2	6.6.2 6.6.9.2	A
7.1-1	Setpoint Calculations for Protective Functions	7.1.6.1	7.1.6.1	B
7.5-1	Post Accident Monitoring	7.5.5	7.5.2 7.5.3.5 7.5.5	A
8.2-1	Offsite Electrical Power	8.2.5	8.2.1 8.2.1.1 8.2.1.2 8.2.1.3 8.2.1.4 8.2.5	A
8.2-2	Technical Interfaces	8.2.5	8.2.1.2.1 8.2.2 8.2.5	A
8.3-1	Grounding and Lightning Protection	8.3.3	8.3.1.1.7 8.3.1.1.8 8.3.3	A
8.3-2	Onsite Electrical Power Plant Procedures	8.3.3	8.3.1.1.2.4 8.3.1.1.6 8.3.2.1.4 8.3.3	A

Table 1.8-202 (Sheet 10 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
9.1-5	Inservice Inspection Program of Cranes	9.1.6.5	9.1.4.4 9.1.5.4 9.1.6	A
9.1-6	Radiation Monitor	9.1.6.6	9.1.4.3.8 9.1.5.3 9.1.6	A
9.1-7	Metamic Monitoring Program	9.1.6.7	9.1.6	H
9.2-1	Potable Water	9.2.11.1	9.2.5.2.1 9.2.5.2.2 9.2.5.3 9.2.5.6 9.2.12.1	A
9.2-2	Waste Water Retention Basins	9.2.11.2	9.2.9.2.1 9.2.9.2.2 9.2.9.5 9.2.12.2	A
9.3-1	Air Systems (NUREG-0933 Issue 43)	9.3.7	9.3.7	A
9.4-1	Ventilation Systems Operations	9.4.12	6.4.4.2 9.4.1.4 9.4.7.4 9.4.12	A

Table 1.8-202 (Sheet 11 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
9.5-1	Qualification Requirements for Fire Protection Program	9.5.1.8.1	9.5.1.6 9.5.1.8 9.5.1.8.1.2 9.5.1.8.2 9.5.1.8.3 9.5.1.8.4 9.5.1.8.5 9.5.1.8.6 9.5.1.8.7 9.5.1.9.1 13.1.1.2.10	A
9.5-2	Fire Protection Analysis Information	9.5.1.8.2	9.5.1.9.2 9A.3.3	A
9.5-3	Regulatory Conformance	9.5.1.8.3	9.5.1.8.1.1 9.5.1.8.8 9.5.1.8.9 9.5.1.9.3 9A.3.3	A
9.5-4	NFPA Exceptions	9.5.1.8.4	9.5.1.8.1.1 9.5.1.9.4	A
9.5-6	Verification of Field Installed Fire Barriers	9.5.1.8.6	9.5.1.8.6 9.5.1.9.6	H

Table 1.8-202 (Sheet 12 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
9.5-8	Establishment of Procedures to Minimize Risk for Fire Areas Breached During Maintenance	9.5.1.8.7	9.5.1.8.1.2 9.5.1.9.7	A
9.5-9	Offsite Interfaces	9.5.2.5.1	9.5.2.5.1	A
9.5-10	Emergency Offsite Communications	9.5.2.5.2	9.5.2.5.2	A
9.5-11	Security Communications	9.5.2.5.3	9.5.2.5.3	A
9.5-13	Fuel Degradation Protection	9.5.4.7.2	9.5.4.5.2 9.5.4.7.2	A
10.1-1	Erosion-Corrosion Monitoring	10.1.3	10.1.3.1 10.1.3.2 10.1.3.3	H
10.2-1	Turbine Maintenance and Inspection	10.2.6	10.2.6	H
10.4-1	Circulating Water Supply	10.4.12.1	10.4.5.2.1 10.4.5.2.2 10.4.5.5 10.4.12.1	A
10.4-2	Condensate, Feedwater and Auxiliary Steam System Chemistry Control	10.4.12.2	10.4.7.2.1 10.4.12.2	A
10.4-3	Potable Water	10.4.12.3	9.2.5.3 10.4.12.3	A

Table 1.8-202 (Sheet 13 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
11.2-1	Liquid Radwaste Processing by Mobile Equipment	11.2.5.1	11.2.1.2.5.2 11.2.5.1	A
11.2-2	Cost Benefit Analysis of Population Doses	11.2.5.2	11.2.3.3 11.2.3.5 11.2.5.2	A
11.3-1	Cost Benefit Analysis of Population Doses	11.3.5.1	11.3.3.4 11.3.5.1	A
11.4-1	Solid Waste Management System Process Control Program	11.4.6	11.4.6	A
11.5-1	Plant Offsite Dose Calculation Manual (ODCM)	11.5.8	11.5.8	A
11.5-2	Effluent Monitoring and Sampling	11.5.8	11.5.1.2 11.5.2.4 11.5.3 11.5.4 11.5.4.1 11.5.4.2 11.5.6.5 11.5.8	A
11.5-3	10 CFR 50, Appendix I	11.5.8	11.2.3.5 11.3.3.4 11.5.8	A

Table 1.8-202 (Sheet 14 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
12.1-1	ALARA and Operational Policies	12.1.3	12.1 12.1.3 Appendix 12AA	A
12.2-1	Additional Contained Radiation Sources	12.2.3	12.2.1.1.10 12.2.3	A
12.3-1	Administrative Controls for Radiological Protection	12.3.5.1	12.3.5.1 Appendix 12AA	A
12.3-2	Criteria and Methods for Radiological Protection	12.3.5.2	12.3.4 12.3.5.2	A
12.3-3	Groundwater Monitoring Program	12.3.5.3	12.3.5.3 Appendix 12AA	A
12.3-4	Record of Operational Events of Interest for Decommissioning	12.3.5.4	12.3.5.4 Appendix 12AA	A
12.5-1	Radiological Protection Organization and Procedures	12.5.5	12.5.5 Appendix 12AA	A
13.1-1	Organizational Structure of Combined License Applicant	13.1.1	13.1.1 13.1.2 13.1.3 13.1.4 Appendix 13AA	A

Table 1.8-202 (Sheet 15 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
13.2-1	Training Program for Plant Personnel	13.2.1	13.2 13.2.1	A
13.3-1	Emergency Planning and Communications	13.3.1	13.3 13.3.6 13.3.7	A
13.3-2	Activation of Emergency Operations Facility	13.3.1	13.3 13.3.6	A
13.4-1	Operational Review	13.4.1	13.4 13.4.1	A
13.5-1	Plant Procedures	13.5.1	13.5 13.5.3	A
13.6-1	Security	13.6	13.6 13.6.1 14.3.2.3.2	A
13.6-5	Cyber Security Program	13.6.1	13.6 13.6.1	H
14.4-1	Organization and Staffing	14.4.1	14.2.2 14.4.1	A
14.4-2	Test Specifics and Procedures	14.4.2	14.4.2	H
14.4-3	Conduct of Test Program	14.4.3	14.4.3	H

VEGP SUP 1.8-2

Table 1.8-202 (Sheet 16 of 18)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
14.4-4	Review and Evaluation of Test Results	14.4.4	14.2.3.2 14.4.4	H
14.4-5	Testing Interface Requirements	14.4.5	14.2.9.4.15 14.2.9.4.22 14.2.9.4.23 14.2.9.4.24 14.2.9.4.25 14.2.9.4.26 14.2.9.4.27 14.2.10.4.29 14.4.5	A
14.4-6	First-Plant-Only and Three-Plant-Only Tests	14.4.6	14.4.6	B
15.0-1	Documentation of Plant Calorimetric Uncertainty Methodology	15.0.15.1	15.0.15 15.0.3.2	H
15.7-1	Consequences of Tank Failure	15.7.6	15.7.6	A
16.1-1	Technical Specification Preliminary Information	16.1	16.1.1	A
16.3-1	Procedure to Control Operability of Investment Protection Systems, Structures and Components	16.3.2	16.3.1 16.3.2	A
17.5-1	Quality Assurance Design Phase	17.5.1	17.1 17.5 17.7	A

Table 1.8-202 (Sheet 17 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
17.5-2	Quality Assurance for Procurement, Fabrication, Installation, Construction and Testing	17.5.2	17.5 17.7	A
17.5-4	Quality Assurance Program for Operations	17.5.4	17.5 17.7	A
17.5-8	Operational Reliability Assurance Program Integration with Quality Assurance Program	17.5.8	17.5 17.7	A
18.2-2	Design of the Emergency Operations Facility	18.2.6.2	9.5.2.2.5 18.2.1.3 18.2.6.2	A
18.6-1	Plant Staffing	18.6.1	13.1.1.4 13.1.3.1 13.1.3.2 18.6 18.6.1	A
18.10-1	Training Program Development	18.10.1	13.1.1.3.1.3.2.2 13.2 18.10 18.10.1	A
18.14-1	Human Performance Monitoring	18.14	18.14	A
19.59.10-1	As-Built SSC HCLPF Comparison to Seismic Margin Evaluation	19.59.10.5	19.59.10.5	H

Table 1.8-202 (Sheet 18 of 18)
COL Item Tabulation

VEGP SUP 1.8-2

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
19.59.10-2	Evaluation of As-Built Plant Versus Design in AP1000 PRA and Site-Specific PRA External Events	19.59.10.5	19.59.10.5	B
19.59.10-3	Internal Fire and Internal Flood Analyses	19.59.10.5	19.59.10.5	H
19.59.10-4	Implement Severe Accident Management Guidance	19.59.10.5	19.59.10.5	H
19.59.10-5	Equipment Survivability	19.59.10.5	19.59.10.5	H
19.59.10-6	Confirm that the Seismic Margin Assessment analysis is applicable to the COL site	19.59.10.5	19.55.6.3 19.59.10.5	A

a) COL Items 1.9-2 and 1.9-3 are un-numbered in the DCD.

VEGP SUP 1.8-3

Table 1.8-203
ESP COL Action Item/FSAR Section Cross-References

ESP COL ITEM	SUBJECT	FSAR SECTION
2.2-1	Hydrazine Hazard from Onsite Storage Tanks	2.2.3.2.3.1
2.2-2	Other Chemical Hazards from Onsite Storage Tanks	2.2.3.2.3.2
2.3-1	Ultimate Heat Sink Design	2.3.1.4
2.4-1	Chelating Agents	11.2.2.1.6
13.6-1	Access Control Measures to Address Existing Rail Spur	13.6.2

VEGP SUP 1.8-5

Table 1.8-204 (Sheet 1 of 2)
ESP Permit Conditions (PC) Cross References

NO.	ESP PERMIT CONDITION	COLA LOCATION
1	The ESP holder shall either remove and replace, or shall improve, the soils directly above the blue bluff marl for soils under or adjacent to Seismic Category 1 structures, to eliminate any liquefaction potential.	Part 10 Appendix B, Safety-Related Backfill ITAAC
2	An applicant for a combined license (COL) referencing this early site permit shall revise the EALs for Unit 3 to reflect the final revision of NEI 07-01.	FSAR Subsection 13.3.8 Part 10, License Condition 4
3	An applicant for a combined license (COL) referencing this early site permit shall revise the EALs for Unit 4 to reflect the final revision of NEI 07-01.	FSAR Subsection 13.3.8 Part 10, License Condition 4
4	An applicant for a combined license (COL) referencing this early site permit shall submit a fully developed EAL scheme for Unit 3 that reflects the completed AP1000 design details, subject to allowable ITAAC.	FSAR Subsection 13.3.8 Part 10, License Condition 4
5	An applicant for a combined license (COL) referencing this early site permit shall submit a fully developed EAL scheme for Unit 4 that reflects the completed AP1000 design details, subject to allowable ITAAC.	FSAR Subsection 13.3.8 Part 10, License Condition 4
6	An applicant for a combined license (COL) referencing this early site permit shall complete a fully developed set of EALs for Unit 3, which are based on in-plant conditions and instrumentation, including onsite and offsite monitoring, and which have been discussed and agreed on by the applicant or licensee and State and local governmental authorities, and shall include the full set of EALs in the COL application. If the EALs are not fully developed, the COL application shall contain appropriate ITAAC for the fully developed set of EALs for Unit 3.	FSAR Subsection 13.3.8 Part 10, License Condition 4

VEGP SUP 1.8-5

Table 1.8-204 (Sheet 2 of 2)
ESP Permit Conditions (PC) Cross References

NO.	ESP PERMIT CONDITION	COLA LOCATION
7	An applicant for a combined license (COL) referencing this early site permit shall complete a fully developed set of EALs for Unit 4, which are based on in-plant conditions and instrumentation, including onsite and offsite monitoring, and which have been discussed and agreed on by the applicant or licensee and State and local governmental authorities, and shall include the full set of EALs in the COL application. If the EALs are not fully developed, the COL application shall contain appropriate ITAAC for the fully developed set of EALs for Unit 4.	FSAR Subsection 13.3.8 Part 10, License Condition 4
8	An applicant for a combined license (COL) referencing this early site permit shall resolve the difference between the VEGP Units 3 and 4 common Technical Support Center (TSC), and the TSC location specified in the AP1000 certified design.	FSAR Subsection 13.3.8 FSAR Section 18.8
9	If a COL or CP application referencing this ESP also references a certified design, the COL or CP applicant may demonstrate compliance with the radiological consequence evaluation factors in 10 CFR 52.79(a)(1) or 10 CFR 50.34(a)(1), respectively, by demonstrating that the site-specific χ/Q values determined in the ESP fall within those evaluated in the approval of the referenced certified design. However, if a COL or CP referencing this ESP does not reference a certified design, the applicant would still need to demonstrate that its source term is bounded by the source term values included in the ESP.	FSAR Table 2.0-201 (Sheet 6) FSAR Table 2.0-202 (Sheets 1-2)

VEGP SUP 1.8-6

Table 1.8-205 (Sheet 1 of 7)
Summary of FSAR Discussions of AP1000 Plant Interfaces

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection(1)
2.1	Envelope of AP1000 plant site related parameters	Site Interface	Site specific parameters	Table 2.0-201 Table 2.0-202
2.2	External missiles from man-made hazards and accidents	Site Interface	Site specific parameters	ESPA SSAR 2.2.2.6 ESPA SSAR 2.2.3.1 2.2.3.2 3.5
2.3	Maximum loads from man-made hazards and accidents	Site Interface	Site specific parameters	ESPA SSAR 2.2.3.1 2.2.3.2
2.4	Limiting meteorological parameters (X/Q) for design basis accidents and for routine releases and other extreme meteorological conditions for the design of systems and components exposed to the environment.	Site Interface	Site specific parameters	Table 2.0-201 Table 2.0-202
2.5	Tornado and operating basis wind loadings	Site Interface	Site specific parameters	Table 2.0-201
2.6	External missiles generated by natural phenomena	Site Interface	Site specific parameters	Table 2.0-201
2.7	Snow, ice and rain loads	Site Interface	Site specific parameters	2.3.1.3.4
2.8	Ambient air temperatures	Site Interface	Site specific parameters	Table 2.0-201
2.9	Onsite meteorological measurement program	Requirement of AP1000	Combined License applicant program	2.3.3.4
2.10	Flood and ground water elevations	Site Interface	Site specific parameters	Table 2.0-201
2.11	Hydrostatic loads on systems, components and structures	Site Interface	Site specific parameters	Table 2.0-201

Table 1.8-205 (Sheet 2 of 7)
Summary of FSAR Discussions of AP1000 Plant Interfaces

VEGP SUP 1.8-6

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection(1)
2.12	Seismic parameters – peak ground acceleration – response spectra – shear wave velocity	Site Interface	Site specific parameters	Table 2.0-201
2.13	Required bearing capacity of foundation materials	Site Interface	Site specific parameters	Table 2.0-201
3.1	Deleted	N/A	N/A	N/A
3.2	Operating procedures to minimize water hammer	Requirement of AP1000	Combined License applicant procedure	10.3.2.2.1 10.4.7.2.1
3.3	Site seismic sensor location and "trigger value"	Requirement of AP1000	Onsite implementation	3.7.4.2.1
3.4	Depth of overburden	Requirement of AP1000	Onsite implementation	ESPA SSAR 2.5.4.5 3.8.5.1
3.5	Depth of embedment	Requirement of AP1000	Onsite implementation	ESPA SSAR 2.5.4.5 3.8.5.1
3.6	Specific depth of waterproofing	Requirement of AP1000	Onsite implementation	ESPA SSAR 2.5.4.5.7 ESPA SSAR 3.8.5.1 3.8.5.1
3.7	Foundation Settlement Monitoring	Requirement of AP1000	Combined License applicant coordination	ESPA SSAR 2.5.4.10.2
3.8	Lateral earth pressure loads	Not an Interface	N/A	N/A
3.9	Preoperational piping vibration test parameters	Not an Interface	N/A	N/A
3.10	Inservice Inspection requirements and locations	Requirement of AP1000	Combined License applicant program	3.9.6 5.2.4 6.6

Table 1.8-205 (Sheet 3 of 7)
Summary of FSAR Discussions of AP1000 Plant Interfaces

VEGP SUP 1.8-6

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection(1)
3.11	Maintenance of preservice and reference test data for inservice testing of pumps and valves	Requirement of AP1000	Combined License applicant program	3.9.6
3.12	Earthquake response procedures	Requirement of AP1000	Combined License applicant program	3.7.4.4
5.1	Steam Generator Tube Surveillance Requirements	Requirement of AP1000	Combined License applicant program	5.4.2.5
6.1	Inservice Inspection requirements for the containment	Requirement of AP1000	Combined License applicant program	6.6
6.2	Off site environmental conditions assumed for Main Control Room and control support area habitability design	AP1000 Interface	Site specific parameter	ESPA SSAR 2.2.3. 2.2.3 6.4
7.1	Listing of all design criteria applied to the design of the I&C systems	Not an Interface	N/A	N/A
7.2	Power required for site service water instrumentation	NNS and Not an Interface	N/A	N/A
7.3	Other provisions for site service water instrumentation	NNS and Not an Interface	N/A	N/A
7.4	Post Accident Monitoring System	NNS	Combined License applicant coordination	7.5.5
8.1	Listing of design criteria applied to the design of the offsite power system	NNS	Combined License applicant coordination	8.1.4.3

Table 1.8-205 (Sheet 4 of 7)
Summary of FSAR Discussions of AP1000 Plant Interfaces

VEGP SUP 1.8-6

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection(1)
8.2	Offsite ac requirements: – Steady-state load; – Inrush kVA for motors; – Nominal voltage; – Allowable voltage regulation; – Nominal frequency; – Allowable frequency fluctuation; – Maximum frequency decay rate; – Limiting under frequency value for RCP	NNS	Combined License applicant coordination	8.2.2
8.3	Offsite transmission system analysis: – Loss of AP1000 or largest unit; – Voltage operating range; – Transient stability must be maintained and the RCP bus voltage must remain above the voltage required to maintain the flow assumed in Chapter 15 analyses for a minimum of three (3) seconds following a turbine trip.; – The protective devices controlling the switchyard breakers are set with consideration given to preserving the plant grid connection following a turbine trip.	NNS	Combined License applicant analysis	8.2.1.2.1 8.2.2 14.2.9.4.23
8.4	Listing of design criteria applied to the design of onsite ac power systems	NNS and Not an Interface	N/A	N/A
8.5	Onsite ac requirements	NNS and Not an Interface	N/A	N/A

Table 1.8-205 (Sheet 5 of 7)
Summary of FSAR Discussions of AP1000 Plant Interfaces

VEGP SUP 1.8-6

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection(1)
8.6	Diesel generator room coordination	NNS and Not an Interface	N/A	N/A
8.7	Listing of design criteria applied to the design of onsite dc power systems	Not an Interface	N/A	N/A
8.8	Provisions of dc power systems to accommodate the site service water system	NNS and Not an Interface	N/A	N/A
9.1	Listing of design criteria applied to the design of portions of the site service water within AP1000	NNS and Not an Interface	N/A	N/A
9.2	Integrated heat load to site service water system	NNS and Not an Interface	N/A	N/A
9.3	Plant cooling water systems parameters	NNS and Not an Interface	N/A	N/A
9.4	Plant makeup water quality limits	NNS	Site specific parameter	9.2.11
9.5	Requirements for location and arrangement of raw and sanitary water systems	NNS	Site implementation	9.2.5 9.2.6 9.2.11
9.6	Ventilation requirements for diesel-generator room	NNS and Not an Interface	N/A	N/A
9.7	Requirements to satisfy fire protection program	AP1000 Interface	Combined License applicant program	9.5.1
9.8	Requirements for location and size of waste water retention basins and associated plant outfall	NNS	Site implementation	9.2.9

Table 1.8-205 (Sheet 6 of 7)
Summary of FSAR Discussions of AP1000 Plant Interfaces

VEGP SUP 1.8-6

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection(1)
11.1	Expected release rates of radioactive material from the Liquid Waste System including: – Location of release points – Effluent temperature – Effluent flow rate – Size and shape of flow orifices	Site Interface	Site specific parameters	11.2
11.2	Expected release rates of radioactive materials from the Gaseous Waste System including: – Location of release points – Height above grade – Height relative to adjacent buildings – Effluent temperature – Effluent flow rate – Effluent velocity – Size and shape of flow orifices	Site Interface	Site specific parameters	11.3
11.3	Expected release rates of radioactive material from the Solid Waste System including: – Location of release points – Material types – Material qualities – Size and shape of material containers	Site Interface	Site specific parameters	11.4.6
11.4	Requirements for offsite sampling and monitoring of effluent concentrations	AP1000 Interface	Combined License applicant program	11.5.3 11.5.8
12.1	Identification of miscellaneous radioactive sources	AP1000 Interface	Combined License applicant program	12.2.1.1.10

Table 1.8-205 (Sheet 7 of 7)
Summary of FSAR Discussions of AP1000 Plant Interfaces

VEGP SUP 1.8-6

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection[1]
13.1	Features that may affect plans for coping with emergencies as specified in 10 CFR 50, Appendix O	AP1000 Interface	Combined License applicant program	13.3
13.2	Physical Security Plan consistent with AP1000 plant	AP1000 Interface	Combined License applicant program	13.6
14.1	Identification of special features to be considered in development of the initial test program	Requirement of AP1000	Combined License applicant program	14
14.2	Maintenance of preoperational test data and inservice inspection baseline data	AP1000 Interface	Combined License applicant program	14
16.1	Administrative requirements associated with reliability information maintenance	AP1000 Interface	Combined License applicant program	16.3
16.2	Administrative requirements associated with the Technical Specifications	Requirement of AP1000	Combined License applicant implementation	16.1
16.3	Site and operator related information associated with the Reliability Assurance Program (D-RAP)	Requirement of AP1000	Combined License applicant program	16.2
18.1	Operating staff consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	18.6
18.2	Operator training consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	18.8 18.10
18.3	Operating Procedures consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	18.8 18.10

Note 1 — This table supplements DCD Table 1.8-1 by providing additional information in the Section or Subsection column. Section/Subsection designations are FSAR unless otherwise noted.

1.9 COMPLIANCE WITH REGULATORY CRITERIA

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

1.9.1 REGULATORY GUIDES

Add the following paragraphs to the end of DCD Subsection 1.9.1:

STD COL 1.9-1 Divisions 2, 3, 6, 7, 9, and 10 of the regulatory guides do not apply to the construction or operational safety considerations and are not addressed in the FSAR.

VEGP COL 1.9-1 Division 4 of the regulatory guides applies to the Environmental Report and the topics are addressed in the Environmental Report. Three Division 4 Regulatory Guides are addressed in Appendix 1AA.

STD COL 1.9-1 Division 5 of the regulatory guides applies to materials and plant protection. As appropriate, the Division 5 regulatory guide topics are addressed in the DCD and plant-specific security plans (i.e., Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Cyber Security Plan).

Applicable Division 8 Regulatory Guides are addressed in Appendix 1AA.

Appendix 1AA provides a discussion of plant specific regulatory guide conformance, addressing new Regulatory Guides and new revisions not addressed by the referenced DCD. Regulatory Guides that are completely addressed by the DCD are not listed.

The following subsections provide a summary discussion of Divisions 1, 4, 5 and 8 of the regulatory guides as applicable to the content of this FSAR, or to the construction and/or operations phases.

1.9.1.1 Division 1 Regulatory Guides - Power Reactors

Add the following paragraphs to the end of DCD Subsection 1.9.1.1:

STD COL 1.9-1 Appendix 1AA provides an evaluation of the degree of compliance with Division 1 regulatory guides as applicable to the content of this FSAR, or to the site-specific design, construction and/or operational aspects. The revisions of the regulatory guides against which the degree of compliance is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. One such general alternative is the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD in order to preserve the finality of the certified design (see Notes at the end of Appendix 1AA). Table 1.9-201 identifies the appropriate regulatory guide to FSAR cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1AA.

Superseded or canceled regulatory guides are not considered in Appendix 1AA or Table 1.9-201.

1.9.1.2 Division 4 Regulatory Guides - Environmental and Siting

Add the following as the first paragraph in DCD Subsection 1.9.1.2:

STD COL 1.9-1 Division 4 of the regulatory guides applies to the Environmental Report and the topics are addressed in the Environmental Report. Appendix 1AA provides an evaluation of the degree of compliance with Division 4 regulatory guides as applicable to the content of this FSAR, or to the site-specific design, construction and/or operational aspects. The revisions of the regulatory guides against which the plant is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. One such general alternative is the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD in order to preserve the finality of the certified design (see Notes at the end of Appendix 1AA). For those regulatory guides applicable, Table 1.9-201 identifies the appropriate FSAR cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1AA.

1.9.1.3 Division 5 Regulatory Guides - Materials and Plant Protection

Add the following as the first paragraph in DCD Subsection 1.9.1.3:

STD COL 1.9-1 Division 5 of the regulatory guides applies to materials and plant protection. Appendix 1AA provides an evaluation of the degree of conformance with Division

5 regulatory guides as applicable to the content of the AP1000 DCD and the plant-specific Cyber Security Plan. The plant-specific physical security plans (i.e., Physical Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan) were developed using the template in NEI 03-12, Revision 6, “Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program],” which was endorsed for use by NRC letter dated April 9, 2009. The plant-specific physical security plans include no substantive deviations from the NRC-endorsed template in NEI 03-12, Revision 6. Therefore, the degree of conformance with Division 5 regulatory guides for the plant-specific physical security plans is consistent with the degree of conformance of NEI 03-12, Revision 6.

1.9.1.4 Division 8 Regulatory Guides - Occupational Health

Add the following paragraphs to the end of DCD Subsection 1.9.1.4:

STD COL 1.9-1 Appendix 1AA provides an evaluation of the degree of compliance with Division 8 regulatory guides as applicable to the content of this FSAR, or to the site-specific design, construction and/or operational aspects. The revisions of the regulatory guides against which the plant is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. One such general alternative is the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD in order to preserve the finality of the certified design (see Notes at the end of Appendix 1AA). For those regulatory guides applicable, Table 1.9-201 identifies the appropriate FSAR cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1AA.

Superseded or canceled regulatory guides are not considered in Appendix 1AA or Table 1.9-201.

1.9.1.5 Combined License Information

Add the following as the first paragraph in DCD Subsection 1.9.1.5:

STD COL 1.9-1 Division 1, 4, 5 and 8 Regulatory Guides applicable to the content of this FSAR, or to the site-specific design, construction and/or operational aspects are listed in Table 1.9-201 and Appendix 1AA.

1.9.2 COMPLIANCE WITH STANDARD REVIEW PLAN (NUREG-0800)

Add the following paragraph to the end of DCD Subsection 1.9.2:

STD SUP 1.9-1 Table 1.9-202 provides the required assessment of conformance with the applicable acceptance criteria and the associated FSAR cross-references.

The design related SRP acceptance criteria addressed by the certified design are identified as such in Table 1.9-202.

1.9.4.1 Review of NRC List of Unresolved Safety Issues and Generic Safety Issues

Add the following paragraphs to the end of DCD Subsection 1.9.4.1:

STD COL 1.9-3 Table 1.9-203 addresses the second un-numbered COL Information Item identified at the end of DCD Table 1.8-2 and listed in Table 1.8-202 as COL Information Item 1.9-3, "Unresolved Safety Issues and Generic Safety Issues." As such, Table 1.9-203 lists those issues on DCD Table 1.9-2 identified by Note "d," which apply to other than design issues, Note "f," which apply either to resolution of Combined License (COL) Information Items or to nuclear power plant operations issues, Note "h," which apply to issues unresolved pending generic resolution at the time of submittal of the AP1000 DCD, and any new Unresolved Safety Issues and Generic Safety Issues that have been included in NUREG-0933 (through supplement 30) since the DCD was developed. Many of these have since been resolved and incorporated into the applicable licensing regulations or guidance (e.g., the standard review plans). These resolved items (as indicated by NUREG-0933) are identified only as "Resolved per NUREG-0933." Many others are not in the list of items in NUREG-0933 Appendix B identified as applicable to new plants. These items are identified only as "Not applicable to new plants." For the remaining items, the table provides the FSAR sections that address the topic.

1.9.4.2.3 New Generic Issues

STD COL 1.9-3 Add the following text in DCD Subsection 1.9.4.2.3., following the AP1000 Position for Issue 185.

Issue 186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants

Discussion:

This issue concerns licensees operating within the regulatory guidelines of Generic Letter 85-11 that may not have taken adequate measures to assess and mitigate the consequences of dropped heavy loads.

FSAR Position:

There are no planned heavy load lifts outside those already described in the DCD. However, over the plant life there may be occasions when heavy loads not presently addressed need to be lifted (i.e. in support of special maintenance/ repairs). For these occasions, special procedures are generated that address the activity. Further discussion is provided in Subsection 9.1.5.3.

Issue 189 Susceptibility of Ice Condenser and Mark III Containments to Early Failure From Hydrogen Combustion During a Severe Accident Description

Discussion:

This issue concerns the early containment failure probability for ice condenser and BWR MARK III containments given the relatively low containment free volume and low containment strength in these designs.

FSAR Position:

The AP1000 design does not have an ice condenser containment or a Mark III containment. Therefore, this issue is not addressed in this FSAR.

Add the following text in DCD Subsection 1.9.4.2.3 following the AP1000 Position for Issue 191.

STD COL 1.9-3 Issue 191 Assessment of Debris Accumulation on PWR Sump Performance (REV. 1)

Discussion:

Results of research on BWR ECCS suction strainer blockage identified new phenomena and failure modes that were not considered in the resolution of Issue A-43. In addition, operating experience identified new contributors to debris and possible blockage of PWR sumps, such as degraded or failed containment paint coatings.

FSAR Position:

The design aspects of this issue are addressed by the DCD. The protective coatings program controls the procurement, application, inspection, and monitoring of Service Level I and Service Level III coatings with the quality assurance features discussed above. The protective coatings program complies with Regulatory Guide 1.54, and is controlled and implemented by administrative procedures. The program is discussed in Subsection 6.1.2.1.6.

Administrative procedures implement the containment cleanliness program. Implementation of the program minimizes the amount of debris that might be left in containment following refueling and maintenance outages. The program is consistent with the containment cleanliness program used in the evaluation discussed in DCD Subsection 6.3.8.2. The program is discussed in Subsection 6.3.8.1.

Issue 196 Boral Degradation

Discussion:

The issue specifically addresses the use of Boral in long-term dry storage casks for spent reactor fuel.

FSAR Position:

Long-term dry storage casks for spent reactor fuel are not used and therefore this issue is not addressed in this FSAR.

1.9.5.1.5 Station Blackout

STD SUP 1.9-3 Add the following text to the end of DCD Subsection 1.9.5.1.5.

Training and procedures to mitigate a 10 CFR 50.63 “loss of all alternating current power” (or station blackout (SBO)) event are implemented in accordance with Sections 13.2 and 13.5, respectively. As recommended by NUMARC 87-00

(Reference 201), the SBO event mitigation procedures 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. The AP1000 is a passive design and does not rely on offsite or onsite ac sources of power for at least 72 hours after an SBO event, as described above.

Restoration from an SBO event will be contingent upon ac power being made available from any one of the transmission lines described in Section 8.2 or any one of the standby diesel generators.

1.9.5.2.15 Severe Accident Mitigation Design Alternatives

Add the following text to the end of DCD Subsection 1.9.5.2.15.

FSAR Position:

VEGP SUP 1.9-2 The severe accident mitigation design alternatives (SAMDA) evaluation for AP1000 contained in DCD Appendix 1B is not incorporated into this FSAR, but is addressed in the ESPA Environmental Report.

1.9.5.5 Operational Experience

Add the following paragraph to the end of DCD Subsection 1.9.5.5.

STD COL 1.9-2 Table 1.9-204 lists the Bulletins and Generic Letters addressed by topical discussion in this FSAR. Table 1.9-204 also lists Bulletins and Generic Letters categorized as part of the first un-numbered COL Information Item identified at the end of DCD Table 1.8-2 and listed in Table 1.8-202 as COL Information Item 1.9-2. Table 1.9-204 provides the appropriate FSAR cross-references for the discussion of the topics addressed by those Bulletins and Generic Letters. Bulletins or Generic Letters issued after those listed in the DCD are also included in Table 1.9-204. Issues identified as “procurement” or “maintenance” or “surveillance” in WCAP-15800 are addressed as part of the scope of the certified design and are not specifically identified in Table 1.9-204. Issues identified as “procedural” in WCAP-15800 are addressed by the procedures discussed in DCD Section 13.5 and are not specifically identified in Table 1.9-204. Other items in WCAP-15800, including the Circulars and Information Notices, are considered to have been adequately addressed based on the guidance identified in Regulatory Guide 1.206 and the NRC Standard Review Plans.

1.9.6 REFERENCES

Add the following text to the end of DCD Subsection 1.9.6.

201. NUMARC 87-00, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors, Revision 1, August 1991.

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(Unless Otherwise Noted)

Table 1.9-201 (Sheet 1 of 16)
Regulatory Guide/FSAR Section Cross-References

		Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
	Division 1 Regulatory Guides		
	1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Rev. 0, March 1971)	16 (TS Bases 3.8.1)
	1.7	Control of Combustible Gas Concentrations in Containment (Rev. 3, March 2007)	DCD discussion only; see DCD Table 1.9-1
VEGP COL 1.9-1	1.8	Qualification and Training of Personnel for Nuclear Power Plants (Rev. 3, May 2000)	12.1 (NEI 07-08A) Appendix 12AA Appendix 12AA (NEI 07-03A) 13.1.1.4 13.1.2.1.1.7 13.1.2.1.1.8 13.1.3.1 13.2 (NEI 06-13A) 16 (TS 5.3.1) 17.5 (QAPD, IV)
	1.11	Instrument Lines Penetrating the Primary Reactor Containment (Rev. 1, March 2010)	DCD discussion only; see DCD Table 1.9-1
	1.12	Nuclear Power Plant Instrumentation for Earthquakes (Rev. 2, March 1997)	3.7.4.1
	1.13	Spent Fuel Storage Facility Design Basis (Rev. 2, March 2007)	16 (TS Bases 3.7.11) 16 (TS Bases 3.7.12)
	1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 3, March 2007)	DCD discussion only; see DCD Table 1.9-1

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Table 1.9-201 (Sheet 2 of 16)
Regulatory Guide/FSAR Section Cross-References

	Regulatory Guides		FSAR Chapter, Section, or Subsection[a]
VEGP COL 1.9-1	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 (Rev. 1, June 1974)	11.5.1.2 11.5.4.1 12.3.4
	1.23	Meteorological Monitoring Programs for Nuclear Power Plants (Pr-1, September 1980)	Note b 2.3.3.4
	1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive - Waste - Containing Components of Nuclear Power Plants (Rev. 4, March 2007)	Note b 5.2.4.1 17.5 (QAPD IV)
	1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 3, August 1985)	14.2.2.2 17.5 (QAPD, II, 17.1) 17.5 (QAPD, IV)
VEGP COL 1.9-1	1.29	Seismic Design Classification (Rev. 3, September 1978, Rev. 4, March 2007)	Note b
	1.29	Seismic Design Classification (Rev. 4, March 2007)	17.5 (QAPD IV)
	1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Rev. 0, August 1972)	Not referenced; see Appendix 1AA
	1.31	Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, April 1978)	6.1.1.2
	1.32	Criteria for Power Systems for Nuclear Power Plants (Rev. 3, March 2004)	16 (TS Bases 3.8.1)
	1.33	Quality Assurance Program Requirements (Operation) (Rev. 2, February 1978)	16 (TS 5.4.1) 17.5 (QAPD, IV)

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Table 1.9-201 (Sheet 3 of 16)
Regulatory Guide/FSAR Section Cross-References

	Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants (Rev. 1, March 2007)	17.5 (QAPD, II, 13.2) 17.5 (QAPD IV)
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, May 1977)	DCD discussion only; see DCD Table 1.9-1
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Rev. 2, September 1977)	DCD discussion only; see DCD Table 1.9-1
1.44	Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)	6.1.1.2
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems (Rev. 0, May 1973)	16 (TS Bases 3.4.7) 16 (TS Bases 3.4.9)
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 (Rev. 3, June 2001)	16 (TS 3.7.6)
1.53	Application of the Single-Failure Criterion to Safety Systems (Rev. 2, November 2003)	DCD discussion only; see DCD Table 1.9-1
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (Rev. 1, July 2000)	1.9.4.2.3 6.1.2.1.6
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 1, March 2007)	DCD discussion only; see DCD Table 1.9-1

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Table 1.9-201 (Sheet 4 of 16)
Regulatory Guide/FSAR Section Cross-References

	Regulatory Guides		FSAR Chapter, Section, or Subsection[a]
VEGP COL 1.9-1	1.59	Design Basis Floods for Nuclear Power Plants (Rev. 2, August 1977)	Note b
	1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)	Note b Table 2.0-201
	1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 1, March 2007)	DCD discussion only; see DCD Table 1.9-1
	1.68	Initial Test Program for Water-Cooled Nuclear Power Plants (Rev. 3, March 2007)	14.2.1 14.2.3 14.2.8 14.2.5.2 16 (TS Bases 3.1.8)
VEGP COL 1.9-1	1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition) (Rev. 3, November 1978)	Note b 1.1.6.1
	1.71	Welder Qualification for Areas of Limited Accessibility (Rev 1, March 2007)	DCD discussion only; see DCD Table 1.9-1
	1.75	Criteria for Independence of Electrical Safety Systems (Rev 3, February 2005)	DCD discussion only; see DCD Table 1.9-1
	1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (Rev. 1, March 2007)	Table 2.0-201
VEGP COL 1.9-1	1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (PR-1, January 2006)	Note b
	1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev 0, May 1974)	16 (TS Bases 3.2.1) 16 (TS Bases 3.2.2) 16 (TS Bases 3.2.4) 16 (TS Bases 3.2.5)

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Regulatory Guide/FSAR Section Cross-References

	Regulatory Guides		FSAR Chapter, Section, or Subsection[a]
VEGP COL 1.9-1	1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (Rev. 1, December 2001)	Note b 2.2.3.2 6.4.3 6.4.4.2 16 (TS Bases 3.7.6) Table 19.58-201
	1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident (Rev. 3, November 2003)	DCD discussion only; see DCD Table 1.9-1
	1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, July 1975)	DCD discussion only; see DCD Table 1.9-1
	1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III (Rev. 33, August 2005)	DCD discussion only; see DCD Table 1.9-1
	1.86	Termination of Operating Licenses for Nuclear Reactors (Rev. 0, June 1974)	Not referenced; see Appendix 1AA
VEGP COL 1.9-1	1.91	Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants (Rev. 1, February 1978)	Note b 2.2.3.2 3.5.1.5 Table 19.58-201
	1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 2, July 2006)	DCD discussion only; see DCD Table 1.9-1
	1.93	Availability of Electric Power Sources (Rev. 0, December 1974)	16 (TS Bases 3.8.1) 16 (TS Bases 3.8.5)
	1.94	Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Rev. 1, April 1976)	Not referenced; see Appendix 1AA

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Table 1.9-201 (Sheet 6 of 16)
Regulatory Guide/FSAR Section Cross-References

		Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
	1.97	Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants (Rev. 4, June 2006)	Not referenced; See Appendix 1AA
	1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following an Accident (Rev. 3, May 1983)	Table 7.5-201 Appendix 12AA 16 (TS Bases 3.3.3)
	1.99	Radiation Embrittlement of Reactor Vessel Materials (Rev. 2, May 1988)	16 (TS Bases 3.4.3)
	1.101	Emergency Response Planning and Preparedness for Nuclear Power Reactors (Rev. 5, June 2005)	Not referenced; see Appendix 1AA
VEGP COL 1.9-1	1.101	Emergency Response Planning and Preparedness for Nuclear Power Reactors (Rev. 4, July 2003)	Note b 9.5.1.8.2.2 Table 9.5-201
	1.102	Flood Protection for Nuclear Power Plants (Rev.1, September 1976)	Note b
	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 (Rev. 1, October 1977)	Note b
	1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (Draft Rev. 0, March 1976)	11.2.3.5.1 11.3.3.4.1
	1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (Rev. 1, July 1977)	Note b

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Regulatory Guide/FSAR Section Cross-References

		Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
	1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors (Rev. 0, April 1976)	Note b
	1.112	Calculation of Releases of Radioactive Materials in Gaseous or Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors (Rev. 1, March 2007)	DCD discussion only; see DCD Table 1.9-1
VEGP COL 1.9-1	1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Rev. 1, April 1977)	Note b
	1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit (Rev. 2, May 1989)	13.1.2.1.3
	1.115	Protection Against Low-Trajectory Turbine Missiles (Rev. 1, July 1977)	3.5.1.3
	1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (Rev. 0-R, May 1977)	Not referenced; see Appendix 1AA
	1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (Rev. 0, August 1976)	16 (TS Bases 3.4.18)
	1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports (Rev. 2, February 2007)	DCD discussion only; see DCD Table 1.9-1
VEGP COL 1.9-1	1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Rev. 1, October 1978)	Note b

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Regulatory Guide/FSAR Section Cross-References

		Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
	1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	DCD discussion only; see DCD Table 1.9-1
	1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	Table 8.1-201 8.3.2.1.4 16 (TS Bases 3.8.1)
	1.130	Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Supports (Rev. 2, March 2007)	DCD discussion only; see DCD Table 1.9-1
VEGP COL 1.9-1	1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 2, October 2003)	Note b
	1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Rev. 1, May 1981)	Not referenced; see Appendix 1AA
	1.134	Medical Evaluation of Licensed Personnel at Nuclear Power Plants (Rev. 3, March 1998)	Not referenced; see Appendix 1AA
	1.135	Normal Water Level and Discharge at Nuclear Power Plants (Rev. 0, September 1977)	DCD discussion only; see DCD Table 1.9-1
VEGP COL 1.9-1	1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants (Rev. 2, December 2003)	Note b
	1.139	Guidance for Residual Heat Removal (Rev. 0, May 1978)	DCD discussion only; see DCD Table 1.9-1
	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 (Rev. 2, June 2001)	9.4.1.4 9.4.7.4 16 (TS Bases 3.9.6)

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Regulatory Guide/FSAR Section Cross-References

		Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
	1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (Rev. 2, November 2001)	11.2.1.2.5.2 11.2.3.6 11.3.3.6 11.4.5 11.4.6.2
VEGP COL 1.9-1	1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (Rev. 1, November 1982)	Note b
	1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1 (Rev. 15, October 2007)	5.2.4 6.6
	1.149	Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations (Rev. 3, October 2001)	13.2 (NEI 06-13A)
	1.150	Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations (Rev. 1, February 1983)	DCD discussion only; see DCD Table 1.9-1
	1.152	Criteria for Use of Computers in Safety Systems of Nuclear Power Plants (Rev. 2, January 2006)	Not referenced; see Appendix 1AA
	1.154	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors (Rev. 0, January 1987)	Not referenced; see Appendix 1AA
	1.155	Station Blackout (Rev. 0, August 1998)	Table 8.1-201
	1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors (Rev. 1, October 2003)	Not referenced; see Appendix 1AA
	1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Rev. 2, March 1997)	3.8.3.7 3.8.4.7 3.8.5.7 17.6 (NEI 07-02A)

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Regulatory Guide/FSAR Section Cross-References

		Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
	1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels (Rev. 0, February 1996)	Not referenced; see Appendix 1AA
	1.163	Performance-Based Containment Leak-Test Program (Rev. 0, September 1995)	6.2.5.1 6.2.5.2.2 16 (TS 5.5.8)
VEGP COL 1.9-1	1.165	Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion (Rev. 0, March 1997)	Note b
	1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post Earthquake Actions (Rev. 0, March 1997)	3.7.4.4
	1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event (Rev. 0, March 1997)	3.7.4.4
	1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 1, February 2004)	DCD discussion only; see DCD Table 1.9-1
	1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis (Rev. 1, November 2002)	Not referenced; see Appendix 1AA
	1.175	An Approach for Plant-Specific, Risk-Informed Decision making: Inservice Testing (Rev. 0, August 1998)	Not referenced; see Appendix 1AA
	1.177	An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications (Rev. 0, August 1998)	16 (TS Bases 3.5.1) 16 (TS Bases 3.7.10)

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Regulatory Guide/FSAR Section Cross-References

		Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
	1.178	An Approach for Plant-Specific Risk-Informed Decision making for Inservice Inspection of Piping (Rev. 1, September 2003)	Not referenced; see Appendix 1AA
	1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors (Rev. 0, January 1999)	Not referenced; see Appendix 1AA
	1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems (Rev. 1, October 2003)	DCD discussion only; see DCD Table 1.9-1
	1.181	Content of Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e) (Rev. 0, September 1999)	Not referenced; see Appendix 1AA
	1.182	Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants (Rev. 0, May 2000)	16 (TS Bases SR 3.0.3) 17.6 (NEI 07-02A)
VEGP COL 1.9-1	1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (Rev. 0, July 2000)	Note b 16 (TS Bases 3.7.5) 16 (TS Bases 3.9.4) 16 (TS Bases 3.9.7)
	1.184	Decommissioning of Nuclear Power Reactors (Rev. 0, July 2000)	Not referenced; see Appendix 1AA
	1.185	Standard Format and Content for Post-shutdown Decommissioning Activities Report (Rev. 0, July 2000)	Not referenced; see Appendix 1AA
	1.186	Guidance and Examples for Identifying 10 CFR 50.2 Design Bases (Rev. 0, December 2000)	Not referenced; see Appendix 1AA
	1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiment (Rev. 0, November 2000)	Not referenced; see Appendix 1AA

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Regulatory Guide/FSAR Section Cross-References

		Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
	1.188	Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses (Rev. 1, September 2005)	Not referenced; see Appendix 1AA
VEGP COL 1.9-1	1.189	Fire Protection for Nuclear Power Plants (Rev. 1, March 2007)	9.5.1.8.1.1 9.5.1.8.2.2 Appendix 9A 13.1.2.1.1.6 17.5 (QAPD III.2)
	1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown (Rev. 0, May 2001)	Not referenced; see Appendix 1AA
	1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code (Rev. 0, June 2003)	3.9.6.3
	1.193	ASME Code Cases Not Approved for Use (Rev 1, August 2005)	Not referenced; see Appendix 1AA
	1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (Rev. 0, June 2003)	2.3.4.3
	1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (Rev. 0, May 2003)	Not referenced; see Appendix 1AA
	1.196	Control Room Habitability at Light-Water Nuclear Power Reactors (Rev. 1, January 2007)	6.4.3
	1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (Rev. 0, May 2003)	DCD discussion only; see DCD Table 1.9-1

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		Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
VEGP COL 1.9-1	1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites (Rev. 0, November 2003)	Note b
	1.199	Anchoring Components and Structural Supports in Concrete (Rev. 0, November 2003)	DCD discussion only; see DCD Table 1.9-1
	1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (Rev. 1, January 2007)	19.59.10.6
	1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance (Rev. 1, May 2006)	Not referenced; see Appendix 1AA
	1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors (Rev. 0, February 2005)	Not referenced; see Appendix 1AA
	1.203	Transient and Accident Analysis Methods (Rev. 0, December 2005)	Not referenced; see Appendix 1AA
	1.204	Guidelines for Lightning Protection of Nuclear Power Plants (Rev. 0, November 2005)	Table 8.1-201
	1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants (Rev. 0, May 2006)	Not referenced; see Appendix 1AA

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Regulatory Guide/FSAR Section Cross-References

		Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
VEGP COL 1.9-1	1.206	Combined License Applications for Nuclear Power Plants (LWR Edition) (Rev. 0, June 2007)	1.1.6.1 Table 1.8-201 1.9.5.5 Table 1.9-201 Table 1.9-202 See Appendix 1AA 2.1 2.2 2.4 Table 8.1-201 Appendix 12AA (NEI 07-03A) 14.3.2.3.2
	1.207	Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors (Rev. 0, March 2007)	Not referenced; see Appendix 1AA
	1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants (Rev. 0, March 2007)	Not referenced; see Appendix 1AA
	Division 4 Regulatory Guides		
VEGP COL 1.9-1	4.2 and Supp. 1	Preparation of Environmental Reports for Nuclear Power Stations (Rev. 2, July 1976; Rev. 0, September 2000)	Note b
	4.7	General Site Suitability Criteria for Nuclear Power Stations (Rev. 2, April 1998)	Note b
	4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) — Effluent Streams and the Environment (Rev. 2, July 2007)	11.5.3

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Table 1.9-201 (Sheet 15 of 16)
Regulatory Guide/FSAR Section Cross-References

Regulatory Guides		FSAR Chapter, Section, or Subsection[a]
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) — Effluent Streams and the Environment (Rev. 1, February 1979)	11.5.1.2 11.5.3 11.5.4 11.5.6.5
Division 5 Regulatory Guides		Note c
Division 8 Regulatory Guides		
8.2	Guide for Administrative Practices in Radiation Monitoring (Rev. 0, February 1973)	12.1 (NEI 07-08A) 12.3.4 Appendix 12AA (NEI 07-03A)
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters (Rev. 0, February 1973)	Appendix 12AA (NEI 07-03A)
8.5	Criticality and Other Interior Evacuation Signals (Rev. 1, March 1981)	Appendix 12AA (NEI 07-03A)
8.6	Standard Test Procedure for Geiger-Muller Counters (Rev. 0, May 1973)	Appendix 12AA (NEI 07-03A)
8.7	Instructions for Recording and Reporting Occupational Radiation Dose Data (Rev. 2, November 2005)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.8	Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable (Rev. 3, June 1978)	12.1 (NEI 07-08A) 12.3.4 Appendix 12AA Appendix 12AA (NEI 07-03A) 13.1.2
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program (Rev. 1, July 1993)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable (Rev. 1-R, May 1977)	12.1 (NEI 07-08A) 12.3.4 Appendix 12AA Appendix 12AA (NEI 07-03A) 13.1.2

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Regulatory Guide/FSAR Section Cross-References

	Regulatory Guides	FSAR Chapter, Section, or Subsection[a]
8.13	Instruction Concerning Prenatal Radiation Exposure (Rev. 3, June 1999)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.15	Acceptable Programs for Respiratory Protection (Rev. 1, October 1999)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.27	Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants (Rev. 0, March 1981)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.28	Audible-Alarm Dosimeters (Rev. 0, August 1981)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.29	Instruction Concerning Risks from Occupational Radiation Exposure (Rev. 1, February 1996)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.34	Monitoring Criteria and Methods To Calculate Occupational Radiation Doses (Rev. 0, July 1992)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.35	Planned Special Exposures (Rev. 0, June 1992)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.36	Radiation Dose to the Embryo/Fetus (Rev. 0, July 1992)	12.1 (NEI 07-08A) Appendix 12AA (NEI 07-03A)
8.38	Control of Access to High and Very High Radiation Areas of Nuclear Plants (Rev. 1, May 2006)	12.1 (NEI 07-08A) Appendix 12AA Table 12AA-201 Appendix 12AA (NEI 07-03A)

a. NEI templates are incorporated by reference. See Table 1.6-201.

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b. This Regulatory Guide is referenced in the ESPA SSAR.

c. Division 5 of the regulatory guides applies to materials and plant protection. As appropriate, the Division 5 regulatory guide topics are addressed in the DCD and plant-specific security plans (i.e., Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Cyber Security Plan).

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Conformance with SRP Acceptance Criteria

		Criteria Section[b]	Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
	1	Introduction and Interfaces, Initial Issuance, 03/2007		N/A	No specific acceptance criteria associated with these general requirements.
VEGP SUP 1.9-2	2.0	Site Characteristics and Site Parameters, Initial Issuance, 03/2007			See Note h.
	2.1.1	Site Location and Description			See Note h.
	2.1.2	Exclusion Area Authority and Control			See Note h.
	2.1.3	Population Distribution			See Note h.
	2.2.1 – 2.2.2	Identification of Potential Hazards in Site Vicinity			See Note h.
	2.2.3	Evaluation of Potential Accidents			See Note h.
	2.3.1	Regional Climatology			See Note h.
	2.3.2	Local Meteorology			See Note h.
	2.3.3	Onsite Meteorological Measurements Programs			See Note h.
	2.3.4	Short-Term Atmospheric Dispersion Estimates for Accident Releases			See Note h.
	2.3.5	Long-Term Atmospheric Dispersion Estimates for Routine Releases			See Note h.

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Conformance with SRP Acceptance Criteria

		Criteria Section[b]	Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
VEGP SUP 1.9-2	2.4.1	Hydrologic Description			See Note h.
	2.4.2	Floods, Rev. 4, 03/2007			See Note h.
	2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers, Rev. 4, 03/2007			See Note h.
	2.4.4	Potential Dam Failures			See Note h.
	2.4.5	Probable Maximum Surge and Seiche Flooding			See Note h.
	2.4.6	Probable Maximum Tsunami Hazards			See Note h.
	2.4.7	Ice Effects			See Note h.
	2.4.8	Cooling Water Canals and Reservoirs			See Note h.
	2.4.9	Channel Diversions			See Note h.
	2.4.10	Flooding Protection Requirements			See Note h.
	2.4.11	Low Water Considerations			See Note h.
	2.4.12	Groundwater			See Note h.
	2.4.13	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters			See Note h.
	2.4.14	Technical Specifications and Emergency Operation Requirements			Acceptable.

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Conformance with SRP Acceptance Criteria

		Criteria Section[b]	Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
VEGP SUP 1.9-2	2.5.1	Basic Geologic and Seismic Information, Rev.4, 03/2007			See Note h.
	2.5.2	Vibratory Ground Motion, Rev. 4, 03/2007			See Note h.
	2.5.3	Surface Faulting, Rev. 4, 03/2007			See Note h.
	2.5.4	Stability of Subsurface Materials and Foundations			See Note h.
	2.5.5	Stability of Slopes			See Note h.
	3.2.1	Seismic Classification, Rev. 2, 03/2007			See Notes d and e.
	3.2.2	System Quality Group Classification, Rev. 2, 03/2007			See Notes d and e.
	3.3.1	Wind Loadings		Acceptable	See Notes d, e, and f.
	3.3.2	Tornado Loadings		Acceptable	See Notes d, e, and f.
	3.4.1	Internal Flood Protection for Onsite Equipment Failures		Acceptable	See Notes d, e, and f.
	3.4.2	Analysis Procedures			See Notes d and e.
	3.5.1.1	Internally Generated Missiles (Outside Containment)			See Notes d and e.
	3.5.1.2	Internally Generated Missiles (Inside Containment)			See Notes d and e.

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	Criteria Section[b]	Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
3.5.1.3	Turbine Missiles		Acceptable	See Notes d, e, and f.
3.5.1.4	Missiles Generated by Tornadoes and Extreme Winds			See Notes d and e.
3.5.1.5	Site Proximity Missiles (Except Aircraft), Rev.4, 03/2007		Acceptable	See Notes d, e, and f.
VEGP SUP 1.9-2 3.5.1.6	Aircraft Hazards			See Note h
3.5.2	Structures, Systems, and Components to be Protected from Externally-Generated Missiles			See Notes d and e.
3.5.3	Barrier Design Procedures			See Notes d and e.
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment			See Notes d and e.
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
3.6.3	Leak-Before-Break Evaluation Procedures, Rev. 1, 03/2007		Acceptable	See Notes d, e, and f.
3.7.1	Seismic Design Parameters			See Notes d and e.
3.7.2	Seismic System Analysis		Acceptable	See Notes d, e, and f.
3.7.3	Seismic Subsystem Analysis			See Notes d and e.

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Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
3.7.4	Seismic Instrumentation, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
3.8.1	Concrete Containment, Rev. 2, 03/2007			See Notes d and e.
3.8.2	Steel Containment, Rev. 2, 03/2007			See Notes d and e.
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments, Rev. 2, 03/2007			See Notes d and e.
3.8.4	Other Seismic Category I Structures, Rev. 2, 03/2007			See Notes d and e.
3.8.5	Foundations, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
3.9.1	Special Topics for Mechanical Components			See Notes d and e.
3.9.2	Dynamic Testing and Analysis of Systems, Structures, and Components			See Notes d and e.
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
3.9.4	Control Rod Drive Systems			See Notes d and e.
3.9.5	Reactor Pressure Vessel Internals			See Notes d and e.
3.9.6	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints		Acceptable	See Notes d, e, and f.

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Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
3.9.7	Risk-Informed Inservice Testing, Rev. 0, 08/1998		N/A	
3.9.8	Risk-Informed Inservice Inspection of Piping, Rev. 0, 09/2003		N/A	
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment			See Notes d and e.
3.11	Environmental Qualification of Mechanical and Electrical Equipment		Acceptable	See Notes d, e, and f.
3.12	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports, Initial Issuance, 03/2007			See Note g.
3.13	Threaded Fasteners - ASME Code Class 1, 2, and 3, Initial Issuance, 03/2007			See Note g.
4.2	Fuel System Design			See Notes d and e.
4.3	Nuclear Design			See Notes d and e.
4.4	Thermal and Hydraulic Design, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
4.5.1	Control Rod Drive Structural Materials			See Notes d and e.
4.5.2	Reactor Internal and Core Support Structure Materials			See Notes d and e.

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Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
4.6	Functional Design of Control Rod Drive System, Rev. 2, 03/2007			See Notes d and e.
5.2.1.1	Compliance with the Codes and Standards Rule, 10 CFR 50.55a		Acceptable	See Notes d, e, and f.
5.2.1.2	Applicable Code Cases			See Notes d and e.
5.2.2	Overpressure Protection			See Notes d and e.
5.2.3	Reactor Coolant Pressure Boundary Materials		Acceptable	See Notes d, e, and f.
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection, Rev. 2, 03/2007			See Notes d and e.
5.3.1	Reactor Vessel Materials, Rev. 2, 03/2007			See Notes d and e.
5.3.2	Pressure-Temperature Limits Upper-Shelf Energy and Pressurized Thermal Shock, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
5.3.3	Reactor Vessel Integrity, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
5.4	Reactor Coolant System Component and Subsystem Design, Rev. 2, 03/2007		N/A	No specific acceptance criteria associated with these general requirements.
5.4.1.1	Pump Flywheel Integrity (PWR), Rev. 2, 03/2007			See Notes d and e.

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Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
5.4.2.1	Steam Generator Materials			See Notes d and e.
5.4.2.2	Steam Generator Program, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
5.4.6	Reactor Core Isolation Cooling System (BWR), Rev. 4, 03/2007		N/A	
5.4.7	Residual Heat Removal (RHR) System, Rev. 4, 03/2007			See Notes d and e.
5.4.8	Reactor Water Cleanup System (BWR)		N/A	
5.4.11	Pressurizer Relief Tank			See Notes d and e.
5.4.12	Reactor Coolant System High Point Vents, Rev. 1, 03/2007			See Notes d and e.
5.4.13	Isolation Condenser System (BWR), Initial Issuance, 03/2007		N/A	
6.1.1	Engineered Safety Features Materials, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
6.1.2	Protective Coating Systems (Paints) – Organic Materials		Acceptable	See Notes d, e, and f.
6.2.1	Containment Functional Design			See Notes d and e.
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments			See Notes d and e.
6.2.1.1.B	Ice Condenser Containments, Rev. 2, 07/1981		N/A	

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Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
6.2.1.1.C	Pressure-Suppression Type BWR Containments, Rev. 7, 03/2007		N/A	
6.2.1.2	Subcompartment Analysis			See Notes d and e.
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)			See Notes d and e.
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures, Rev. 2, 03/2007			See Notes d and e.
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies			See Notes d and e.
6.2.2	Containment Heat Removal Systems, Rev. 5, 03/2007			See Notes d and e.
6.2.3	Secondary Containment Functional Design			See Notes d and e.
6.2.4	Containment Isolation System			See Notes d and e.
6.2.5	Combustible Gas Control in Containment		Acceptable	See Notes d, e, and f.
6.2.6	Containment Leakage Testing		Acceptable	See Notes d, e, and f.
6.2.7	Fracture Prevention of Containment Pressure Boundary, Rev. 1, 03/2007			See Notes d and e.
6.3	Emergency Core Cooling System		Acceptable	See Notes d, e, and f.

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Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
6.4	Control Room Habitability System		Acceptable	See Notes d, e, and f.
6.5.1	ESF Atmosphere Cleanup Systems			See Notes d and e.
6.5.2	Containment Spray as a Fission Product Cleanup System, Rev. 4, 03/2007			See Notes d and e.
6.5.3	Fission Product Control Systems and Structures			See Notes d and e.
6.5.4	Ice Condenser as a Fission Product Cleanup System, Rev. 3, 12/1988		N/A	
6.5.5	Pressure Suppression Pool as a Fission Product Cleanup System, Rev. 1, 03/2007		N/A	
6.6	Inservice Inspection and Testing of Class 2 and 3 Components, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
6.7	Main Steam Isolation Valve Leakage Control System (BWR), Rev. 2, 07/1981		N/A	
7	Instrumentation and Controls –Overview of Review Process, Rev. 5, 03/2007			See Notes d and e.
Appendix 7.0-A	Review Process for Digital Instrumentation and Control Systems, Rev. 5, 03/2007			See Notes d and e.

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Conformance with SRP Acceptance Criteria

Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
7.1	Instrumentation and Controls –Introduction, Rev. 5, 03/2007			See Notes d and e.
7.1-T Table 7-1	Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety, Rev. 5, 03/2007			See Notes d and e.
Appendix 7.1-A	Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety, Rev. 5, 03/2007			See Notes d and e.
Appendix 7.1-B	Guidance for Evaluation of Conformance to IEEE Std 279, Rev. 5, 03/2007			See Notes d and e.
Appendix 7.1-C	Guidance for Evaluation of Conformance to IEEE Std 603, Rev. 5, 03/2007			See Notes d and e.
Appendix 7.1-D	Guidance for Evaluation of the Application of IEEE Std 7-4.3.2 Initial Issuance 03/2007			See Notes d and e.
7.2	Reactor Trip System, Rev. 5, 03/2007			See Notes d and e.
7.3	Engineered Safety Features Systems, Rev. 5, 03/2007			See Notes d and e.
7.4	Safe Shutdown Systems, Rev. 5, 03/2007			See Notes d and e.
7.5	Information Systems Important to Safety, Rev. 5, 03/2007			See Notes d and e.

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Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
7.6	Interlock Systems Important to Safety, Rev. 5, 03/2007			See Notes d and e.
7.7	Control Systems, Rev. 5, 03/2007			See Notes d and e.
7.8	Diverse Instrumentation and Control Systems, Rev. 5, 03/2007			See Notes d and e.
7.9	Data Communication Systems, Rev. 5, 03/2007			See Notes d and e.
8.1	Electric Power – Introduction		N/A	No specific acceptance criteria associated with these general requirements.
8.2	Offsite Power System, Rev. 4, 03/2007		Acceptable	See Notes d, e, and f
8.3.1	A-C Power Systems (Onsite)		Acceptable	See Notes d, e, and f.
8.3.2	D-C Power Systems (Onsite)		Acceptable	See Notes d, e, and f.
8.4	Station Blackout, Initial Issuance, 03/2007			See Note g.
9.1.1	Criticality Safety of Fresh and Spent Fuel Storage and Handling			See Notes d and e.
9.1.2	New and Spent Fuel Storage, Rev. 4, 03/2007			See Notes d and e.
9.1.3	Spent Fuel Pool Cooling and Cleanup System, Rev. 2, 03/2007			See Notes d and e.

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Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
9.1.4	Light Load Handling System (Related to Refueling)		Acceptable	See Notes d, e, and f.
9.1.5	Overhead Heavy Load Handling Systems, Rev. 1, 03/2007		Acceptable	See Notes d, e, and f.
9.2.1	Station Service Water System, Rev. 5, 03/2007		Acceptable	See Notes d, e, and f.
9.2.2	Reactor Auxiliary Cooling Water Systems, Rev. 4, 03/2007			See Notes d and e.
9.2.4	Potable and Sanitary Water Systems			See Notes d and e.
9.2.5	Ultimate Heat Sink		Acceptable	See Notes d, e, and f.
9.2.6	Condensate Storage Facilities		Acceptable	See Notes d, e, and f.
9.3.1	Compressed Air System, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
9.3.2	Process and Post-accident Sampling Systems			See Notes d and e.
9.3.3	Equipment and Floor Drainage System			See Notes d and e.
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)			See Notes d and e.
9.3.5	Standby Liquid Control System (BWR)		N/A	
9.4.1	Control Room Area Ventilation System		Acceptable	See Notes d, e, and f.
9.4.2	Spent Fuel Pool Area Ventilation System			See Notes d and e.

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Criteria Section[(b)]		Reference Criteria	FSAR Position[(c)]	Comments/Summary of Exceptions
9.4.3	Auxiliary and Radwaste Area Ventilation System			See Notes d and e.
9.4.4	Turbine Area Ventilation System			See Notes d and e.
9.4.5	Engineered Safety Feature Ventilation System			See Notes d and e.
9.5.1	Fire Protection Program, Rev. 5, 03/2007		Acceptable	See Notes d, e, and f.
9.5.2	Communications Systems		Acceptable	See Notes d, e, and f.
9.5.3	Lighting Systems			See Notes d and e.
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System		Acceptable	See Notes d, e, and f.
9.5.5	Emergency Diesel Engine Cooling Water System			See Notes d and e.
9.5.6	Emergency Diesel Engine Starting System			See Notes d and e.
9.5.7	Emergency Diesel Engine Lubrication System			See Notes d and e.
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System			See Notes d and e.
10.2	Turbine Generator		Acceptable	See Notes d, e, and f.
10.2.3	Turbine Rotor Integrity, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
10.3	Main Steam Supply System, Rev. 4, 03/2007		Acceptable	See Notes d, e, and f.

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Table 1.9-202 (Sheet 15 of 27)[(a)]
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	Criteria Section[(b)]	Reference Criteria	FSAR Position[(c)]	Comments/Summary of Exceptions
10.3.6	Steam and Feedwater System Materials		Acceptable	See Notes d, e, and f.
10.4.1	Main Condensers			See Notes d and e.
10.4.2	Main Condenser Evacuation System		Acceptable	See Notes d, e, and f.
10.4.3	Turbine Gland Sealing System			See Notes d and e.
10.4.4	Turbine Bypass System			See Notes d and e.
10.4.5	Circulating Water System		Acceptable	See Notes d, e, and f.
10.4.6	Condensate Cleanup System			See Notes d and e.
10.4.7	Condensate and Feedwater System, Rev. 4, 03/2007		Acceptable	See Notes d, e, and f.
10.4.8	Steam Generator Blowdown System (PWR)			See Notes d and e.
10.4.9	Auxiliary Feedwater System (PWR)			See Notes d and e.
11.1	Source Terms			See Notes d and e.
VEGP SUP 1.9-2 11.2	Liquid Waste Management System		Acceptable	See Notes d, e, f, and h.
11.3	Gaseous Waste Management System			See Notes d, e, f, and h.
11.4	Solid Waste Management System		Acceptable	See Notes d, e, and f.
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems, Rev. 4, 03/2007		Acceptable	See Notes d, e, and f.

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	Criteria Section[b]	Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
12.1	Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable		Exception	See Notes d, e, and f. An exception is taken to following the guidance of RG 1.206 to address RG 8.20, 8.25, and RG 8.26. NUREG-1736, Final Report (published 2001) lists RG 8.20 and RG 8.26 as “outdated” and recommends the methods of RG 8.9 R1. RG 8.25 states it is not applicable to nuclear facilities licensed under 10 CFR Part 50, and, by extension, to 10 CFR Part 52. An exception is taken to RG 8.8 C.3.b. RG 1.16 C.1.b (3) data is no longer reported. Reporting per C.1.b (2) is also no longer required.
12.2	Radiation Sources		Exception	See Notes d, e, and f. A general description of miscellaneous sealed sources related to radiography is provided in FSAR text. Other requested details are maintained on-site for NRC review and audit upon their procurement.
12.3 – 12.4	Radiation Protection Design Features		Acceptable	See Notes d, e, and f.

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	Criteria Section[b]	Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
12.5	Operational Radiation Protection Program		Acceptable	See Notes d, e, and f.
13.1.1	Management and Technical Support Organization, Rev. 5, 03/2007		Exception	See Notes d, e, and f. Design and construction responsibilities are not defined in numbers. The experience requirements of corporate staff are set by corporate policy and not provided here in detail, however the experience level of the corporate staff, as discussed Subsections 13.1.1, 13.1.1.1, and Appendix 13AA, in the area of nuclear plant development, construction, and management establishes that the applicant has the necessary capability and staff to ensure that design and construction of the facility will be performed in an acceptable manner.

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Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
				Resumes and/or other documentation of qualification and experience of initial appointees to appropriate management and supervisory positions are available for NRC after position vacancies are filled.
13.1.2 –13.1.3	Operating Organization, Rev. 6, 03/2007		Exception	See Notes d, e, and f. The SRP requires resumes of personnel holding plant managerial and supervisory positions to be included in the FSAR. Current industry practice is to have the resumes available for review by the regulator when requested but not be kept in the FSAR. Additionally, at time of COLA, most positions are unfilled.

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	Criteria Section[b]	Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
13.2.1	Reactor Operator Requalification Program; Reactor Operator Training		Exception	See Notes d, e, and f. SRP requires meeting the guidance of NUREG-0711. NEI 06-13A, Template for an Industry Training Program Description, which is incorporated by reference in FSAR 13.2, does not address meeting the guidance of NUREG-0711. NEI 06-13A, is approved by NRC to meet the regulatory requirements for the FSAR description of the Training Program. SRP requires meeting the guidance of Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations" RG 1.149 is not addressed in NEI 06-13A. Level of detail is consistent with NEI 06-13A.
13.2.2	Non-Licensed Plant Staff Training		Exception	See Notes d, e, and f. Level of detail is consistent with NEI 06-13A.
VEGP SUP 1.9-2 13.3	Emergency Planning			See Notes d, e, f, and h.
13.4	Operational Programs		Acceptable	See Notes d, e, and f.

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	Criteria Section[b]	Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
13.5.1.1	Administrative Procedures – General, Initial Issuance, 03/2007		Exception	The procedure development schedule is addressed in the COL application (not in the SAR as requested by this SRP).
13.5.2.1	Operating and Emergency Operating Procedures, Rev. 2, 03/2007		Exception	See Notes d, e, and f. Procedures are generally identified in this section by topic, type, or classification in lieu of the specific title and represent general areas of procedural coverage.
13.6	Physical Security		Acceptable	See Security Plan developed in accordance with NEI 03-12.
13.6.1	Physical Security - Combined License Review Responsibilities, Initial Issuance, 03/2007		Acceptable	See Security Plan developed in accordance with NEI 03-12
13.6.2	Physical Security - Design Certification, Initial Issuance, 03/2007			See notes d and e.
VEGP SUP 1.9-2 13.6.3	Physical Security - Early Site Permit, Initial Issuance, 03/2007			See Note h.
14.2	Initial Plant Test Program - Design Certification and New License Applicants		Exception	See Notes d, e, and f. The level of detail is consistent with DCD section content addressing nonsafety-related systems.

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Conformance with SRP Acceptance Criteria

Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
14.2.1	Generic Guidelines for Extended Power Uprate Testing Programs, Initial Issuance, 08/2006		N/A	No power uprate is sought.
14.3	Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		Acceptable	
14.3.1	[Reserved]			
14.3.2	Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007			See Notes d and e.
14.3.3	Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007			See Notes d and e.
14.3.4	Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007			See Notes d and e.
14.3.5	Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007			See Notes d and e.
14.3.6	Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007			See Notes d and e.
14.3.7	Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		Acceptable	See Notes d, e, and f.

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Conformance with SRP Acceptance Criteria

		Criteria Section[b]	Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
	14.3.8	Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007			See Notes d and e.
	14.3.9	Human Factors Engineering - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007			See Notes d and e.
VEGP SUP 1.9-2	14.3.10	Emergency Planning - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		Exception	See Subsection 14.3.2.3.1.
	14.3.11	Containment Systems - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007			See Notes d and e.
	14.3.12	Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		Acceptable	See Notes d, e, and f.
	15	Introduction —Transient and Accident Analysis			See Notes d and e.
	15.0.1	Radiological Consequence Analyses Using Alternative Source Terms, Rev. 0, 07/2000			See Notes d and e.
	15.0.2	Review of Transient and Accident Analysis Method, Rev. 0, 12/2005			See Notes d and e.
	15.0.3	Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors, Initial Issuance, 03/2007			See Notes d and e.

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Conformance with SRP Acceptance Criteria

Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
15.1.1 – 15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve, Rev. 2, 03/2007			See Notes d and e.
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)			See Notes d and e.
15.2.1 – 15.2.5	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed), Rev. 2, 03/2007			See Notes d and e.
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries, Rev. 2, 03/2007			See Notes d and e.
15.2.7	Loss of Normal Feedwater Flow, Rev. 2, 03/2007			See Notes d and e.
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR), Rev. 2, 03/2007			See Notes d and e.
15.3.1 – 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions, Rev. 2, 03/2007			See Notes d and e.
15.3.3 – 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break			See Notes d and e.

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Conformance with SRP Acceptance Criteria

Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition			See Notes d and e.
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power			See Notes d and e.
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)			See Notes d and e.
15.4.4 –15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate, Rev. 2, 03/2007			See Notes d and e.
15.4.6	Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR), Rev. 2, 03/2007			See Notes d and e.
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position, Rev. 2, 03/2007			See Notes d and e.
15.4.8	Spectrum of Rod Ejection Accidents (PWR)			See Notes d and e.
15.4.8.A	Radiological Consequences of a Control Rod Ejection Accident (PWR) , Rev. 1, 07/1981			See Notes d and e.
15.4.9	Spectrum of Rod Drop Accidents (BWR)		N/A	

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Conformance with SRP Acceptance Criteria

		Criteria Section[(b)]	Reference Criteria	FSAR Position[(c)]	Comments/Summary of Exceptions
	15.5.1 – 15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory, Rev. 2, 03/2007			See Notes d and e.
	15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve, Rev. 2, 03/2007			See Notes d and e.
	15.6.5	Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary			See Notes d and e.
	15.8	Anticipated Transients Without Scram, Rev. 2, 03/2007			See Notes d and e.
	15.9	Boiling Water Reactor Stability, Initial Issuance, 03/2007		N/A	
	16	Technical Specifications, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
	16.1	Risk-informed Decision Making: Technical Specifications, Rev. 1, 03/2007		N/A	This SRP applies to the Technical Specifications change process.
VEGP SUP 1.9-2	17.1	Quality Assurance During the Design and Construction Phases, Rev. 2, 07/1981		Acceptable	See Notes d, e, and f.
	17.2	Quality Assurance During the Operations Phase, Rev. 2, 07/1981			See Notes d and e.

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Conformance with SRP Acceptance Criteria

		Criteria Section[b]	Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
	17.3	Quality Assurance Program Description, Rev. 0, 08/1990			See Notes d and e.
	17.4	Reliability Assurance Program (RAP), Initial Issuance, 03/2007			See Notes d and e.
	17.5	Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants, Initial Issuance, 03/2007		Acceptable	See Notes d, e, and f. This section covers the requirements of SRP Section 17.5 through reference to Quality Assurance Program Description which is maintained separately and developed in accordance with NEI 06-14A.
VEGP SUP 1.9-2	17.6	Maintenance Rule, Rev 1, 08/2007		Acceptable	Content developed in accordance with NEI 07-02A
	18.0	Human Factors Engineering, Rev. 2, 03/2007		Acceptable	See Notes d, e, and f.
	19.0	Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors, Rev. 2, 06/2007		Acceptable	See Notes d, e, and f.
	19.1	Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Rev. 2, 06/2007		Acceptable	See Notes d, e, and f.

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Conformance with SRP Acceptance Criteria

Criteria Section[b]		Reference Criteria	FSAR Position[c]	Comments/Summary of Exceptions
19.2	Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance, Initial Issuance, 06/2007			See Note g.

a) This table is provided as a one-time aid to facilitate NRC review. This table becomes historical information and need not be updated

b) If no revision or date is specified, it is Rev. 3, 03/2007.

c) Consult the AP1000 Design Control Document (DCD) Appendix 1A and Appendix 1AA to determine extent of conformance with Regulatory Guides (except Regulatory Guide 1.206).

d) Conformance with a previous revision of this SRP is documented in AP1000 Design Control Document (Section 1.9.2 and WCAP-15799)

e) Conformance with the design aspects of this SRP is as stated in the AP1000 DCD.

f) Conformance with the plant or site-specific aspects of this SRP is as stated under "FSAR Position."

g) This SRP is not applicable to the AP1000 certified design.

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h) Conformance with RS-002 and NUREG-0800 criteria contained in referenced ESPA. Refer to ESPA SSAR Table 1-2.

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Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
TMI Action Plan Items			
I.A.1.1	Shift Technical Advisor	f	Resolved per NUREG-0933
I.A.1.2	Shift Supervisor Administrative Duties	f	Resolved per NUREG-0933
I.A.1.3	Shift Manning	f	Resolved per NUREG-0933
I.A.1.4	Long-Term Upgrading	f	Resolved per NUREG-0933
I.A.2.1(1)	Qualifications - Experience	f	Resolved per NUREG-0933
I.A.2.1(2)	Immediate Upgrading of RO & SRO Training and Qualifications, Training	f	Resolved per NUREG-0933
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	f	Resolved per NUREG-0933
I.A.2.3	Administration of Training Programs	f	Resolved per NUREG-0933
I.A.2.4	NRR Participation in Inspector Training	d	Not applicable to new plants
I.A.2.6(1)	Revise Regulatory Guide 1.8	f	Resolved per NUREG-0933
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	f	Resolved per NUREG-0933
I.A.3.5	Establish Statement of Understanding with INPO and DOE	d	Not applicable to new plants
I.A.4.1(2)	Interim Changes in Training Simulators	f	Resolved per NUREG-0933
I.A.4.2(1)	Research on Training Simulators	f	Resolved per NUREG-0933
I.A.4.2(2)	Upgrade Training Simulator Standards	f	Resolved per NUREG-0933
I.A.4.2(3)	Regulatory Guide on Training Simulators	f	Resolved per NUREG-0933
I.A.4.2(4)	Review Simulators for Conformance to Criteria	f	Resolved per NUREG-0933
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	d	Not applicable to new plants

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STD COL 1.9-3 Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.A.4.4	Feasibility Study of NRC Engineering Computer	d	Not applicable to new plants
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error	d	Not applicable to new plants
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	d	Not applicable to new plants
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	d	Not applicable to new plants
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	d	Not applicable to new plants
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	d	Not applicable to new plants
I.B.2.1(3)	Follow-up on Completed Maintenance Work Orders to Ensure Proper Testing and Return to Service	d	Not applicable to new plants
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	d	Not applicable to new plants
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	d	Not applicable to new plants
I.B.2.1(6)	Observe Routine Maintenance	d	Not applicable to new plants
I.B.2.1(7)	Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses	d	Not applicable to new plants
I.B.2.2	Resident Inspector at Operating Reactors	d	Not applicable to new plants
I.B.2.3	Regional Evaluations	d	Not applicable to new plants
I.B.2.4	Overview of Licensee Performance	d	Not applicable to new plants
I.C.1(1)	Small Break LOCAs	f	Resolved per NUREG-0933
I.C.1(2)	Inadequate Core Cooling	f	Resolved per NUREG-0933
I.C.1(3)	Transients and Accidents	f	Resolved per NUREG-0933
I.C.2	Shift and Relief Turnover Procedures	f	Resolved per NUREG-0933

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Table 1.9-203 (Sheet 3 of 15)
Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.C.3	Shift Supervisor Responsibilities	f	Resolved per NUREG-0933
I.C.4	Control Room Access	f	Resolved per NUREG-0933
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	f	Resolved per NUREG-0933
I.C.7	NSSS Vendor Review of Procedures	f	Resolved per NUREG-0933
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	f	Resolved per NUREG-0933
I.D.5(5)	Disturbance Analysis Systems	d	Not applicable to new plants
I.D.6	Technology Transfer Conference	d	Not applicable to new plants
I.E.1	Office for Analysis and Evaluation of Operational Data	d	Not applicable to new plants
I.E.2	Program Office Operational Data Evaluation	d	Not applicable to new plants
I.E.3	Operational Safety Data Analysis	d	Not applicable to new plants
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	d	Not applicable to new plants
I.E.5	Nuclear Plant Reliability Data Systems	d	Not applicable to new plants
I.E.6	Reporting Requirements	d	Not applicable to new plants
I.E.7	Foreign Sources	d	Not applicable to new plants
I.E.8	Human Error Rate Analysis	d	Not applicable to new plants
I.F.2(6)	Increase the Size of Licensees' QA Staff	f	Resolved per NUREG-0933
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	f	Resolved per NUREG-0933
I.G.1	Training Requirements	f	Resolved per NUREG-0933

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Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.G.2	Scope of Test Program	f	Resolved per NUREG-0933
II.B.4	Training for Mitigating Core Damage	f	Resolved per NUREG-0933
II.B.5(1)	Behavior of Severely Damaged Fuel	d	Not applicable to new plants
II.B.5(2)	Behavior of Core Melt	d	Not applicable to new plants
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structures	d	Not applicable to new plants
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	f	Resolved per NUREG-0933
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	d	Resolved per NUREG-0933
II.E.6.1	Test Adequacy Study	d	Resolved per NUREG-0933
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	d	Not applicable to new plants
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	d	Not applicable to new plants
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	d	Not applicable to new plants
II.J.1.2	Modify Existing Vendor Inspection Program	d	Not applicable to new plants
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	d	Not applicable to new plants
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	d	Not applicable to new plants
II.J.2.1	Reorient Construction Inspection Program	d	Not applicable to new plants
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	d	Not applicable to new plants
II.J.2.3	Assign Resident Inspectors to All Construction Sites	d	Not applicable to new plants

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STD COL 1.9-3 Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.J.3.1	Organization and Staffing to Oversee Design and Construction	f	Not applicable to new plants
II.J.4.1	Revise Deficiency Reporting Requirements	f	Resolved per NUREG-0933
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	f	Resolved per NUREG-0933
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	f	Resolved per NUREG-0933
II.K.1(4)	Review Operating Procedures and Training Instructions	f	Resolved per NUREG-0933
II.K.1(5)	Safety-Related Valve Position Description	f	Resolved per NUREG-0933
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	f	Resolved per NUREG-0933
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	f	Resolved per NUREG-0933
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	f	Resolved per NUREG-0933
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	f	Resolved per NUREG-0933
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	f	Resolved per NUREG-0933
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	f	Resolved per NUREG-0933
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	f	Resolved per NUREG-0933
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	f	Resolved per NUREG-0933
II.K.1(16)	Implement Procedures That Identify PZR PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	f	Resolved per NUREG-0933

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Table 1.9-203 (Sheet 6 of 15)
Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	f	Resolved per NUREG-0933
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	f	Resolved per NUREG-0933
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	f	Resolved per NUREG-0933
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	f	Resolved per NUREG-0933
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	f	Resolved per NUREG-0933
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	f	Resolved per NUREG-0933
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	f	Resolved per NUREG-0933
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	f	Resolved per NUREG-0933
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	f	Resolved per NUREG-0933
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	f	Resolved per NUREG-0933
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	f	Not applicable to new plants
III.A.2.1(1)	Publish Proposed Amendments to the Rules	d	Resolved per NUREG-0933
III.A.2.1(2)	Conduct Public Regional Meetings	d	Not applicable to new plants
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	d	Not applicable to new plants
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	d	Resolved per NUREG-0933
III.A.2.2	Development of Guidance and Criteria	d	Resolved per NUREG-0933

Table 1.9-203 (Sheet 7 of 15)
STD COL 1.9-3 Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
III.A.3.3	Communications	d	Resolved per NUREG-0933
III.C.1(1)	Review Publicly Available Documents	d	Not applicable to new plants
III.C.1(2)	Recommend Publication of Additional Information	d	Not applicable to new plants
III.C.1(3)	Program of Seminars for News Media Personnel	d	Not applicable to new plants
III.C.2(1)	Develop Policy and Procedures for Dealing With Briefing Requests	d	Not applicable to new plants
III.C.2(2)	Provide Training for Members of the Technical Staff	d	Not applicable to new plants
III.D.2.4(2)	Place 50 TLDs Around Each Site	d	Not applicable to new plants
III.D.2.6	Independent Radiological Measurements	d	Not applicable to new plants
III.D.3.2(1)	Amend 10 CFR 20	d	Not applicable to new plants
III.D.3.2(2)	Issue a Regulatory Guide	d	Not applicable to new plants
III.D.3.2(3)	Develop Standard Performance Criteria	d	Not applicable to new plants
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	d	Not applicable to new plants
III.D.3.3	In-Plant Radiation Monitoring	COL Item 12.3-2	12.3.4, Appendix 12AA
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	d	Not applicable to new plants
III.D.3.5(2)	Investigate Methods of Obtaining Employee Health Data by Nonlegislative Means	d	Not applicable to new plants
III.D.3.5(3)	Revise 10 CFR 20	d	Not applicable to new plants
IV.A.1	Seek Legislative Authority	d	Not applicable to new plants
IV.A.2	Revise Enforcement Policy	d	Not applicable to new plants

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STD COL 1.9-3 Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	d	Not applicable to new plants
IV.D.1	NRC Staff Training	d	Not applicable to new plants
IV.E.1	Expand Research on Quantification of Safety Decision-Making	d	Not applicable to new plants
IV.E.2	Plan for Early Resolution of Safety Issues	d	Not applicable to new plants
IV.E.3	Plan for Resolving Issues at the CP Stage	d	Not applicable to new plants
IV. E.4	Resolve Generic Issues by Rulemaking	d	Not applicable to new plants
IV.G.1	Develop a Public Agenda for Rulemaking	d	Not applicable to new plants
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	d	Not applicable to new plants
IV.G.3	Improve Rulemaking Procedures	d	Not applicable to new plants
IV.G.4	Study Alternatives for Improved Rulemaking Process	d	Not applicable to new plants
IV.H.1	NRC Participation in the Radiation Policy Council	d	Not applicable to new plants
V.A.1	Develop NRC Policy Statement on Safety	d	Not applicable to new plants
V.B.1	Study and Recommend, as Appropriate, Elimination of Nonsafety Responsibilities	d	Not applicable to new plants
V.C.1	Strengthen the Role of Advisory Committee on Reactor Safeguards	d	Not applicable to new plants
V.C.2	Study Need for Additional Advisory Committees	d	Not applicable to new plants
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	d	Not applicable to new plants
V.D.1	Improve Public and Intervenor Participation in the Hearing Process	d	Not applicable to new plants
V.D.2	Study Construction-During-Adjudication Rules	d	Not applicable to new plants

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Table 1.9-203 (Sheet 9 of 15)
Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
V.D.3	Reexamine Commission Role in Adjudication	d	Not applicable to new plants
V.D.4	Study the Reform of the Licensing Process	d	Not applicable to new plants
V.E.1	Study the Need for TMI-Related Legislation	d	Not applicable to new plants
V.F.1	Study NRC Top Management Structure and Process	d	Not applicable to new plants
V.F.2	Reexamine Organization and Functions of the NRC Offices	d	Not applicable to new plants
V.F.3	Revise Delegations of Authority to Staff	d	Not applicable to new plants
V.F.4	Clarify and Strengthen the Respective Roles of Chairman, Commission, and Executive Director for Operations	d	Not applicable to new plants
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	d	Not applicable to new plants
V.G.1	Achieve Single Location, Long-Term	d	Not applicable to new plants
V.G.2	Achieve Single Location, Interim	d	Not applicable to new plants
Task Action Plan Items			
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	COL Item 5.4-1	5.4.2.5
A-19	Digital Computer Protection System	d	Not applicable to new plants
A-20	Impacts of the Coal Fuel Cycle	d	Not applicable to new plants
A-23	Containment Leak Testing	COL Item 6.2-1	6.2.5.1
A-27	Reload Applications	d	Not applicable to new plants
B-1	Environmental Technical Specifications	d	Not applicable to new plants
B-2	Forecasting Electricity Demand	d	Not applicable to new plants
B-11	Subcompartment Standard Problems	d	Not applicable to new plants

Table 1.9-203 (Sheet 10 of 15)
STD COL 1.9-3 Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
B-13	Marviken Test Data Evaluation	d	Not applicable to new plants
B-20	Standard Problem Analysis	d	Not applicable to new plants
B-25	Piping Benchmark Problems	d	Not applicable to new plants
B-27	Implementation and Use of Subsection NF	d	Not applicable to new plants
B-28	Radionuclide/Sediment Transport Program	d	Not applicable to new plants
B-29	Effectiveness of Ultimate Heat Sinks	d	Not applicable to new plants
B-30	Design Basis Floods and Probability	d	Not applicable to new plants
B-33	Dose Assessment Methodology	d	Not applicable to new plants
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors	d	Not applicable to new plants
B-37	Chemical Discharges to Receiving Waters	d	Not applicable to new plants
B-42	Socioeconomic Environmental Impacts	d	Not applicable to new plants
B-43	Value of Aerial Photographs for Site Evaluation	d	Not applicable to new plants
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	d	Not applicable to new plants
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	d	Not applicable to new plants
B-59	(N-1) Loop Operation in BWRs and PWRs	d	Not applicable to new plants
B-64	Decommissioning of Reactors	f	Resolved per NUREG-0933.
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	d	Not applicable to new plants

Table 1.9-203 (Sheet 11 of 15)
STD COL 1.9-3 Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
C-4	Statistical Methods for ECCS Analysis	d	Not applicable to new plants
C-5	Decay Heat Update	d	Not applicable to new plants
C-6	LOCA Heat Sources	d	Not applicable to new plants
New Generic Issues			
43.	Reliability of Air Systems	f, j	Resolved per NUREG-0933.
59.	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable	d	Not applicable to new plants
67.2.1	Integrity of Steam Generator Tube Sleeves	d	Not applicable to new plants
67.5.1	Reassessment of Radiological Consequences	d	Not applicable to new plants
67.5.2	Reevaluation of SGTR Design Basis	d	Not applicable to new plants
67.10.0	Supplement Tube Inspections	d	Not applicable to new plants
99.	RCS/RHR Suction Line Valve Interlock on PWRs	f	Resolved per NUREG-0933.
111.	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	d	Not applicable to new plants
112.	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	d	Not applicable to new plants
118.	Tendon Anchorage Failure	f	Resolved per NUREG-0933.
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	d	Not applicable to new plants
119.3	Decoupling the OBE from the SSE	d	Not applicable to new plants
119.4	BWR Piping Materials	d	Not applicable to new plants
119.5	Leak Detection Requirements	d	Not applicable to new plants

Table 1.9-203 (Sheet 12 of 15)
STD COL 1.9-3
Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
128.	Electrical Power Reliability	h (High)	Resolved per NUREG-0933.
130.	Essential Service Water Pump Failures at Multiplant Sites	f	See DCD Subsection 1.9.4, item 130
133.	Update Policy Statement on Nuclear Plant Staff Working Hours	d	Not applicable to new plants
136.	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	d	Not applicable to new plants
139.	Thinning of Carbon Steel Piping in LWRs	d	Not applicable to new plants
146.	Support Flexibility of Equipment and Components	d	Not applicable to new plants
147.	Fire-Induced Alternate Shutdown Control Room Panel Interactions	d	Not applicable to new plants
148.	Smoke Control and Manual Fire-Fighting Effectiveness	d	Not applicable to new plants
155.2	Establish Licensing Requirements For Non-Operating Facilities	d	Not applicable to new plants
156	Systematic Evaluation Program	f	Not applicable to new plants
156.6.1	Pipe Break Effects on Systems and Components	High	The AP1000 is a new plant that takes the effects of a pipe break into account and therefore issue 156.6.1 is not applicable.
163	Multiple Steam Generator Tube Leakage	h (High)	See DCD Subsection 1.9.4.2.3, item 163
168	Environmental Qualification Of Electrical Equipment	f	Not applicable to new plants
178	Effect Of Hurricane Andrew On Turkey Point	d	Not applicable to new plants
180	Notice Of Enforcement Discretion	d	Not applicable to new plants

STD COL 1.9-3

Table 1.9-203 (Sheet 13 of 15)
Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
181	Fire Protection	d	Not applicable to new plants
183	Cycle-Specific Parameter Limits In Technical Specifications	d	Not applicable to new plants
184	Endangered Species	d	Not applicable to new plants
185	Control of Recriticality following Small-Break LOCA in PWRs	h	Not applicable to new plants
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	Continue	1.9.4.2.3 9.1.5.3
189	Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident Description	Continue	Not applicable to the AP1000.
191	Assessment Of Debris Accumulation On PWR Sump Performance	h (High)	See DCD Subsections 6.3.2.2.7 and 1.9.4.2.3, item 191
199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	Issue to be Prioritized by NRC in the Future	2.5
Human Factors Issues			
HF1.1	Shift Staffing	f	13.1.2.1.4 18.6
HF2.1	Evaluate Industry Training	d	Not applicable to new plants
HF2.2	Evaluate INPO Accreditation	d	Not applicable to new plants
HF2.3	Revise SRP Section 13.2	d	Not applicable to new plants
HF3.1	Develop Job Knowledge Catalog	d	Not applicable to new plants
HF3.2	Develop License Examination Handbook	d	Not applicable to new plants
HF3.5	Develop Computerized Exam System	d	Not applicable to new plants

STD COL 1.9-3

Table 1.9-203 (Sheet 14 of 15)
Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
HF4.2	Procedures Generation Package Effectiveness Evaluation	d	Not applicable to new plants
HF7.1	Human Error Data Acquisition	d	Not applicable to new plants
HF7.2	Human Error Data Storage and Retrieval	d	Not applicable to new plants
HF7.3	Reliability Evaluation Specialist Aids	d	Not applicable to new plants
HF7.4	Safety Event Analysis Results Applications	d	Not applicable to new plants
Chernobyl Issues			
CH1.1A	Symptom-Based EOPs	d	Not applicable to new plants
CH1.1B	Procedure Violations	d	Not applicable to new plants
CH1.2A	Test, Change, and Experiment Review Guidelines	d	Not applicable to new plants
CH1.2B	NRC Testing Requirements	d	Not applicable to new plants
CH1.3A	Revise Regulatory Guide 1.47	d	Not applicable to new plants
CH1.4A	Engineered Safety Feature Availability	d	Not applicable to new plants
CH1.4B	Technical Specification Bases	d	Not applicable to new plants
CH1.4C	Low Power and Shutdown	d	Not applicable to new plants
CH1.5	Operating Staff Attitudes Toward Safety	d	Not applicable to new plants
CH1.6A	Assessment of NRC Requirements on Management	d	Not applicable to new plants
CH1.7A	Accident Management	d	Not applicable to new plants
CH2.1A	Reactivity Transients	d	Not applicable to new plants
CH2.3B	Contamination Outside Control Room	d	Not applicable to new plants

STD COL 1.9-3

Table 1.9-203 (Sheet 15 of 15)
Listing Of Unresolved Safety Issues And Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
CH2.3C	Smoke Control	d	Not applicable to new plants
CH2.3D	Shared Shutdown Systems	d	Not applicable to new plants
CH2.4A	Firefighting With Radiation Present	d	Not applicable to new plants
CH3.1A	Containment Performance	d	Not applicable to new plants
CH3.2A	Filtered Venting	d	Not applicable to new plants
CH4.3A	Ingestion Pathway Protective Measures	d	Not applicable to new plants
CH4.4A	Decontamination	d	Not applicable to new plants
CH4.4B	Relocation	d	Not applicable to new plants
CH5.1A	Mechanical Dispersal in Fission Product Release	d	Not applicable to new plants
CH5.1B	Stripping in Fission Product Release	d	Not applicable to new plants
CH5.2A	Steam Explosions	d	Not applicable to new plants
CH6.1B	Structural Graphite Experiments	d	Not applicable to new plants
CH6.2	Assessment	d	Not applicable to new plants

Notes (from DCD Table 1.9-2):

(d) Issue is not a design issue (Environmental, Licensing, or Regulatory Impact Issue; or covered in an existing NRC program).

(f) Issue is not an AP1000 design certification issue. Issue is applicable to current operating plants or is programmatic in nature.

(h) Issue is unresolved pending generic resolution (for example, prioritized as High, Medium, or possible resolution identified).

(j) The AP600 DSER (Draft NUREG-0612) identified this item as required to be discussed.

STD COL 1.9-2
(Unless Otherwise Noted)

Table 1.9-204 (Sheet 1 of 6)
Generic Communications Assessment

	Number	Title	Comment
	BULLETIN		
	80-06	Engineered Safety Feature (ESF) Reset Controls (3/80)	See Note a.
	80-10	Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment (5/80)	Appendix 12AA
VEGP COL 1.9-2	80-15	Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power (6/80)	9.5.2.2.5 9.5.2.5.1
	88-11	Pressurizer Surge Line Thermal Stratification	3.9.3.1.2
	02-01	Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity	5.2.4 See Note a.
	02-02	Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs	5.2.4 See Note a.
	03-01	Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors	6.3 See Note a.
	03-02	Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity	5.2.4.3 See Note a.
	03-03	Potentially Defective 1-inch Valves for Uranium Hexafluoride Cylinders	N/A
	03-04	Rebaselining of Data in the Nuclear Materials Management and Safeguards System	N/A One time report.

STD COL 1.9-2
(Unless Otherwise Noted)

Table 1.9-204 (Sheet 2 of 6)
Generic Communications Assessment

Number	Title	Comment
04-01	Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors	See Note a.
05-01	Material Control and Accounting at Reactors and Wet Spent Fuel Storage Facilities	13.5.2.2.9
05-02	Emergency Preparedness and Response Actions for Security-Based Events	13.3
GENERIC LETTERS		
80-22	Transmittal of NUREG-0654 “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans” (3/80)	13.3
80-26	Qualifications of Reactor Operators (3/80)	13.2 18.10
80-51	On-Site Storage Of Low-Level Waste (6/90)	11.4.6
80-55	Possible Loss of Hotline With Loss Of Off-Site Power	See Bulletin 80-15
80-77	Refueling Water Level (8/80)	16.1 See Note a.
80-094	Emergency Plan (11/80)	13.3
80-099	Technical Specification Revisions for Snubber Surveillance (11/80)	Snubbers no longer in generic Tech Specs See Note a.
80-108	Emergency Planning (12/80)	13.3
81-02	Analysis, Conclusions and Recommendations Concerning Operator Licensing (1/81)	13.2

STD COL 1.9-2
(Unless Otherwise Noted)

Table 1.9-204 (Sheet 3 of 6)
Generic Communications Assessment

	Number	Title	Comment
	81-10	Post-TMI Requirements for the Emergency Operations Facility (2/81)	13.3
	81-38	Storage of Low-Level Radioactive Waste at Power Reactor Sites (11/81)	11.4.6
	81-40	Qualifications of Reactor Operators (12/81)	13.1 13.2
	82-02	Commission Policy on Overtime (2/82)	16.1
	82-04	Use of INPO See-in Program (3/82)	13.1 13.5
VEGP COL 1.9-2	82-12	Nuclear Power Plant Staff Working Hours (6/82)	13.1.2.1.2 13.1.2.1.3 13.1.2.1.4
	82-13	Reactor Operator and Senior Reactor Operator Examinations (6/82)	For information only.
	82-18	Reactor Operator and Senior Reactor Operator Requalification Examinations (10/82)	13.2
	83-06	Certificates and Revised Format For Reactor Operator and Senior Reactor Operator Licenses (1/83)	13.2
	83-11	Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions (2/83)	13.1 See Note a.
	83-12	Issuance of NRC FORM 398 - Personal Qualifications Statement - Licensee (2/83)	13.2
	83-17	Integrity of the Requalification Examinations for Renewal of Reactor Operator and Senior Reactor Operator Licenses (4/83)	13.1
	83-22	Safety Evaluation of "Emergency Response Guidelines" (6/83)	18.9

STD COL 1.9-2
(Unless Otherwise Noted)

Table 1.9-204 (Sheet 4 of 6)
Generic Communications Assessment

Number	Title	Comment
83-40	Operator Licensing Examination (12/83)	13.2
84-10	Administration of Operating Tests Prior to Initial Criticality (10 CFR 55.25) (4/84)	13.2
84-14	Replacement and Requalification Training Program (5/84)	13.2
84-17	Annual Meeting to Discuss Recent Developments Regarding Operator Training, Qualifications, and Examinations (7/84)	Administrative
84-20	Scheduling Guidance for Licensee Submittals of Reloads That Involve Unreviewed Safety Questions (8/84)	13.5
85-04	Operating Licensing Examinations (1/85)	Administrative
85-05	Inadvertent Boron Dilution Events (1/85)	13.5
85-14	Commercial Storage At Power Reactor Sites Of Low Level Radioactive Waste Not Generated By The Utility (8/85)	Administrative
85-18	Operator Licensing Examinations (9/85)	Administrative
85-19	Reporting Requirements On Primary Coolant Iodine Spikes (9/85)	16.1
86-14	Operator Licensing Examinations (8/86)	Administrative
87-14	Operator Licensing Examinations (8/87)	Administrative
88-05	Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants (3/88)	5.2.4 See Note a.
88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment (8/88)	9.3.7
88-18	Plant Record Storage on Optical Disk (10/88)	17

STD COL 1.9-2
(Unless Otherwise Noted)

Table 1.9-204 (Sheet 5 of 6)
Generic Communications Assessment

Number	Title	Comment
89-07	Power Reactors Safeguards Contingency Planning for Surface Vehicle Bombs (4/89)	13.6
89-07 S1	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	13.6
89-08	Erosion/Corrosion-Induced Pipe Wall Thinning	10.1.3.1
89-12	Operator Licensing Examination (7/89)	13.2
89-15	Emergency Response Data System (8/89)	9.5.2 13.3
89-17	Planned Administrative Changes to the NRC Operator Licensing Written Examination Process (9/89)	N/A
91-14	Emergency Telecommunications (9/91)	9.5.2 13.3
91-16	Licensed Operators and Other Nuclear Facility Personnel Fitness for Duty (10/91)	13.7
92-01	Reactor Vessel Structural Integrity (1/92)	5.3.2.6.3
93-01	Emergency Response Data System Test Program	13.3
93-03	Verification of Plant Records	17
96-02	Reconsideration of Nuclear Power Plant Security Requirements Associated with an Internal Threat (2/96)	13.6
03-01	Control Room Habitability	6.4 See Note a.
04-01	Requirements for Steam Generator Tube Inspections	5.4.2.5 16.1 See Note a.

STD COL 1.9-2
(Unless Otherwise Noted)

Table 1.9-204 (Sheet 6 of 6)
Generic Communications Assessment

Number	Title	Comment
04-02	Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors	6.3.8.1 See Note a.
06-01	Steam Generator Tube Integrity and Associated Technical Specifications	5.4.2.5 16.1 See Note a.
06-02	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	8.2.1.1 8.2.2 See Note a.
06-03	Potentially Nonconforming Hemyc and MT Fire Barrier Configurations	9.5.1.8 See Note a.
07-01	Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients.	17.6 See Note a.

(a) The design aspects of this topic are as stated in the AP1000 DCD.

Add the following section after DCD Section 1.9.

1.10 NUCLEAR POWER PLANTS TO BE OPERATED ON MULTI-UNIT SITES

STD SUP 1.10-1 The certification for the AP1000 is for a single unit. Dual siting of AP1000 is achievable, provided that the centerlines of the units are sufficiently separated. The primary consideration in setting this separation distance is the space needed to support plant construction via the use of a heavy-lift crane.

Security controls during construction and operation are addressed in the Physical Security Plan.

Management and administrative controls are established to identify potential hazards to structures, systems, and components (SSCs) of an operating unit as a result of construction activities at a unit under construction. Controls within this section are not required unless there is an operating unit on the site, i.e., a unit with fuel loaded into the reactor vessel. Advance notification, scheduling and planning allow site management to implement interim controls to reduce the potential for impact to SSCs.

This section presents 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 assessment includes:

- Identification of potential construction activity hazards
- Identification of SSCs important to safety and limiting conditions for operation (LCOs) for the operating unit
- Identification of potentially impacted SSCs and LCOs
- Identification of applicable managerial and administrative controls

1.10.1 POTENTIAL CONSTRUCTION ACTIVITY HAZARDS

VEGP SUP 1.10-1 The power blocks for Units 3 and 4 have a minimum separation of at least 800 feet between plant centerlines.

STD SUP 1.10-1 Construction activities may include site exploration, grading, clearing, and installation of drainage and erosion-control measures; boring, drilling, dredging, pile driving and excavating; transportation, storage and warehousing of equipment; and construction, erection, and fabrication of new facilities.

Construction activities and their representative hazards to an operating unit are shown in Table 1.10-201.

1.10.2 POTENTIALLY IMPACTED SSCS AND LIMITING CONDITIONS FOR OPERATION

The construction activities described above were reviewed for possible impact to operating unit SSCs important to safety.

VEGP SUP 1.10-1

- VEGP Unit 1 and Unit 2 SSCs important to safety are described in Chapter 3 of the Updated Final Safety Analysis Report (UFSAR).
- VEGP Unit 1 and Unit 2 LCOs are located in Appendix A of the Unit 1 and Unit 2 Operating Licenses (Technical Specifications).
- Unit 3 and Unit 4 SSCs important to safety are described in FSAR Chapter 3.
- As indicated in Chapter 16, the LCOs for Units 3 and 4 are located in Part 4 of the COL Application.

STD SUP 1.10-1

The initial assessment consisted of a review of individual SSCs and LCOs to determine whether an item is applicable, or may be eliminated due to either examination or being internal and specific to an operating unit. The assessment identified the SSCs that could reasonably be expected to be impacted by construction activities unless administrative and managerial controls are established. The results of the assessment are presented in Table 1.10-202.

Periodic assessment during construction is addressed in Appendix 13AA, Subsection 13AA.1.1.1.1.8.

1.10.3 MANAGERIAL AND ADMINISTRATIVE CONTROLS

To eliminate or mitigate construction hazards that could potentially impact operating unit SSCs important to safety, specific managerial and administrative controls have been identified as shown in Table 1.10-203.

Although not all of the managerial and administrative construction controls are necessary to protect the operating unit, the identified controls are applied to any operating unit as a conservative measure. This conservative approach provides reasonable assurance of protecting the identified SSCs from potential construction hazards and preventing the associated LCOs specified in the operating unit Technical Specifications from being exceeded as a result of construction activities, as discussed below.

The majority of the operating unit SSCs important to safety are contained and protected within safety-related structures. The managerial controls protect these internal SSCs from postulated construction hazards by maintaining the integrity and design basis of the safety-related structures and foundations. Heavy load drop controls, crane boom failure standoff requirements, ground vibration controls and construction generated missile(s) control are examples of managerial controls that provide this protection.

Other managerial controls support maintaining offsite power, control of hazardous materials and gases, and protection of cooling water supplies and safety system instrumentation. These managerial controls prevent or mitigate external construction impacts that could affect SSCs important to safety. These controls also prevent or mitigate unnecessary challenges to safety systems caused by plant construction hazards, such as disruption of offsite transmission lines or impact to plant cooling water supplies.

The above discussed controls to eliminate or mitigate construction hazards that could potentially impact operating unit SSCs important to safety are in place when there is an operating nuclear unit on the site. Additional controls may be established during construction as addressed in Appendix 13AA, Subsection 13AA.1.1.1.1.8.

STD SUP 1.10-1

Table 1.10-201 (Sheet 1 of 3)
Potential Hazards from Construction Activities

CONSTRUCTION ACTIVITY HAZARD	POTENTIAL IMPACT
Site Exploration, Grading, Clearing, Installation of Drainage and Erosion Control Measures	• Overhead Power Lines • Transmission Towers • Underground Conduits, Piping, Tunnels, etc. • Site Access and Egress • Drainage Facilities and Structures • Onsite Transportation Routes • Slope Stability • Soil Erosion and Local Flooding • Construction-Generated Dust and Equipment Exhausts • Encroachment on Plant Control Boundaries • Encroachment on Structures and Facilities
Boring, Drilling, Pile Driving, Dredging, Demolition, Excavation	• Underground Conduits, Piping, Tunnels, etc. • Foundation Integrity • Structural Integrity • Slope Stability • Erosion and Turbidity Control • Groundwater and Groundwater Monitoring Facilities • Dewatering Structures, Systems and Components • Nearby Structures, Systems and Components • Vibratory Ground Motion

STD SUP 1.10-1

Table 1.10-201 (Sheet 2 of 3)
Potential Hazards from Construction Activities

CONSTRUCTION ACTIVITY HAZARD	POTENTIAL IMPACT
Equipment Movement, Material Delivery, Vehicle Traffic	• Overhead Power Lines • Transmission Towers • Underground Conduits, Piping, Tunnels • Crane Load Drops • Crane or Crane Boom Failures • Vehicle Accidents • Rail Car Derailments
Equipment and Material Laydown, Storage, Warehousing	• Releases of Flammable, Hazardous or Toxic Materials • Wind-Generated, Construction-Related Debris and Missiles
General Construction, Erection, Fabrication	• Physical Integrity of Structures, Systems and Components • Adjacent or Nearby Structures, Systems and Components • Instrumentation and Control Systems and Components • Electrical Systems and Components·Cooling Water Systems and Components • Waste Heat Environmental Controls and Parameters • Radioactive Waste Release Points and Parameters • Abandonment of Structures, Systems or Components • Relocation of Structures, Systems or Components • Removal of Structures, Systems or Components

STD SUP 1.10-1

Table 1.10-201 (Sheet 3 of 3)
Potential Hazards from Construction Activities

CONSTRUCTION ACTIVITY HAZARD	POTENTIAL IMPACT
Connection, Integration, Testing	• Instrumentation and Control Systems and Components • Electrical and Power Systems and Components • Cooling Water Systems and Components

STD SUP 1.10-1

Table 1.10-202 (Sheet 1 of 2)
Hazards During Construction Activities

CONSTRUCTION HAZARD	IMPACTED SSCs
Impact on Overhead Power Lines	• Offsite Power System
Impact on Transmission Towers	• Offsite Power Systems
Impact on Utilities, Underground Conduits, Piping, Tunnels, Tanks	• Fire Protection System • Service Water System[1]
Impact of Construction-Generated Dust and Equipment Exhausts	• Control Room Emergency HVAC Systems[1] • Diesel Generators
Impact of Vibratory Ground Motion	• Offsite Power System • Onsite Power Systems • Instrumentation and Seismic Monitors
Impact of Crane or Crane Boom Failures	• Safety-Related Structures
Impact of Releases of Flammable, Hazardous or Toxic Materials	• Control Room Emergency HVAC Systems[1]
Impact of Wind-Generated, Construction-Related Debris and Missiles	• Safety-Related Structures • Control Room Emergency HVAC Systems[1]
Impact on Electrical Systems and Components	• Offsite Power System • Onsite Power Systems
Impact on Cooling Water Systems and Components	• Service Water System[1] • Ultimate Heat Sink[1]
Impact on Radioactive Waste Release Points and Parameters	• Gaseous and Liquid Radioactive Waste Management Systems

STD SUP 1.10-1

Table 1.10-202 (Sheet 2 of 2)
Hazards During Construction Activities

CONSTRUCTION HAZARD	IMPACTED SSCs
Impact of Relocation of Structures, Systems or Components	• Fire Protection System • Service Water System[1]
Impact of Site Groundwater Depression and Dewatering	• Safety-Related Structures and Foundations
Impact of Equipment Delivery and Heavy Equipment Delivery	• Safety-Related Structures and Foundations
Impact of Local Flooding	• Safety-related structures, systems, and components (SSCs)

[1] Not applicable to AP1000 operating units.

STD SUP 1.10-1

Table 1.10-203 (Sheet 1 of 3)
Managerial and Administrative Construction Controls

CONSTRUCTION HAZARDS TO SSCs	MANAGERIAL CONTROL
Impact on Transmission Power Lines and Offsite Power Lines	• Safe standoff clearance distances are established for transmission power lines, including verification of standoff distance for modules, the reactor vessel and other equipment to be transported beneath energized electric lines to meet minimum standoff clearance requirements. • Physical warning or caution barriers and signage are erected along transport routes.
Impact on Transmission Towers	• Establish controls or physical barriers to avoid equipment collisions with electric transmission support towers
Impact on Utilities, Underground Conduits, Piping, Tunnels, Tanks	• Grading, excavation, and pile driving require location and identification of equipment or underground structures that must be relocated, removed, or left in place and protected prior to the work activity.
Impact of Construction-Generated Dust and Equipment Exhausts	• Fugitive dust and dust generation is controlled. Potentially affected system air intakes and filters are periodically monitored.
Impact of Vibratory Ground Motion	• Construction administrative procedures, methods, and controls are implemented to prevent exceeding ground vibration and instrumentation limit settings.
Impact of Crane or Crane Boom Failures	• Construction standoff distance controls prevent heavy load impacts from crane boom failures and crane load drops. Drop analyses may be substituted if minimum standoff distances are not practical.

STD SUP 1.10-1

Table 1.10-203 (Sheet 2 of 3)
Managerial and Administrative Construction Controls

CONSTRUCTION HAZARDS TO SSCs	MANAGERIAL CONTROL
Impact of Releases of Flammable, Hazardous or Toxic Materials and Missile Generation	• Environmental, safety and health controls limit transport, storage, quantities, type and use of flammable, hazardous, toxic materials and compressed gasses. Construction safety and storage controls maintain potential missile generation events from compressed gasses within the operating unit design basis.
Impact of Wind-Generated, Construction-Related Debris and Missiles	• Administrative controls address equipment, material storage and transport during high winds or high wind warnings. • Plant procedures are followed during severe weather conditions which may call for power reduction or shut down.
Impact on Electrical Systems and Components	• Affected operating unit electrical systems and components within the construction area are identified and isolated or relocated or otherwise protected.
Impact on Cooling Water Systems and Components	• Transport of heavy load equipment over buried cooling water piping is prohibited without evaluation.
Impact on Radioactive Waste Release Points and Parameters	• Engineering evaluation and managerial controls are implemented, as necessary, to prevent radioactive releases beyond the established limits due to construction activity.
Impact of Relocation of Structures, Systems or Components	• Administrative controls identify SSCs that require relocation. Temporary or permanent design changes are implemented if necessary.
Impact of Equipment Delivery and Heavy Equipment Delivery	• Rail transport speed limits and maximum rail loading weights onsite are established. • General equipment and heavy equipment movement controls and limitations are established.

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Table 1.10-203 (Sheet 3 of 3)
Managerial and Administrative Construction Controls

CONSTRUCTION HAZARDS TO SSCs	MANAGERIAL CONTROL
Impact of Local Flooding	• Site grading and drainage provisions consider potential flooding impacts from local intense precipitation
Impact of Site Groundwater Dewatering	• Administrative controls address groundwater level monitoring

APPENDIX 1A
CONFORMANCE WITH REGULATORY GUIDES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD COL 1.9-1 Appendix 1AA is provided to supplement the information in DCD Appendix 1A.

APPENDIX 1AA
CONFORMANCE WITH REGULATORY GUIDES

STD COL 1.9-1
(Unless Otherwise Noted)

Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
DIVISION 1 — Power Reactors			
Regulatory Guide 1.7, Rev. 3, 03/07 – Control of Combustible Gas Concentrations in Containment			
Conformance of the design aspects with Revision 2 of the Regulatory Guide is as stated in the DCD. Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.2		Conforms	
C.4		Conforms	
Regulatory Guide 1.8, Rev. 3, 5/00 – Qualification and Training of Personnel for Nuclear Power Plants			
C.1		Conforms	
C.2	Section 4 of ANSI/ANS-3.1-1993	Exception	Not able to meet Regulatory Guide 1.8, Rev. 3 qualification requirements for licensed personnel prior to operations.
Regulatory Guide 1.11, Rev. 1, 3/10 – Instrument Lines Penetrating the Primary Reactor Containment			
Conformance with the design aspects is as stated in the DCD. This guidance is completely within the scope of the DCD.			
Regulatory Guide 1.12, Rev. 2, 3/97 – Nuclear Power Plant Instrumentation for Earthquakes			
Conformance of the design aspects is as stated in the DCD. Conformance for programmatic and/or operational aspects is documented below.			
C.3		Conforms	
C.8		Conforms	
Regulatory Guide 1.13, Rev. 2, 03/07 - Spent Fuel Storage Facility Design Basis			
Conformance of the design aspects with Revision 1 of the Regulatory Guide is as stated in the DCD. Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.7		Conforms	

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 1.20, Rev. 3, 3/07 – Comprehensive Vibration Assessment Program For Reactor Internals During Preoperational and Initial Startup Testing			
Conformance with Revision 2 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.			
Regulatory Guide 1.21, Rev. 1, 6/74 – 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			
Conformance of the design aspects is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.1		Conforms	
C.3-C.5		Conforms	
C.6		Conforms	
C.7-C.14		Conforms	
VEGP COL 1.9-1 **Regulatory Guide 1.23, Pr-1, September 1980 – Meteorological Monitoring Programs for Nuclear Power Plants**			
General		Note a	
Regulatory Guide 1.26, Rev. 4, 3/07 – Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants			
Conformance with Revision 3 of the Regulatory Guide for DCD scope of work is as stated in the DCD. Conformance with Revision 4 of this Regulatory Guide for remaining scope is documented below.			
General		Conforms	
Regulatory Guide 1.28, Rev. 3, 8/85 – Quality Assurance Program Requirements (Design and Construction)			
Conformance for DCD scope of work is as stated in the DCD. Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 1.29, Rev. 4, 3/07 – Seismic Design Classification			
Conformance with Revision 3 of the Regulatory Guide for DCD scope of work is as stated in the DCD. Conformance with Revision 4 of this Regulatory Guide for remaining scope is documented below.			
C.4		Conforms	
VEGP COL 1.9-1 **Regulatory Guide 1.29, Rev. 3, 9/78 and Rev. 4, 3/07 – Seismic Design Classification**			
		Note a	
Regulatory Guide 1.30, Rev. 0, 8/72 – Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment			
Conformance for DCD scope of work is as stated in the DCD. Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Regulatory Guide 1.32, Rev. 3, 03/04 – Criteria for Power Systems for Nuclear Power Plants			
Conformance of the design aspects with Revision 2 of the Regulatory Guide is as stated in the DCD. Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 1.33, Rev. 2, 2/78 – Quality Assurance Program Requirements (Operation)			
C.1		Conforms	
C.2		Clarification	See separate conformance statement for each identified Regulatory Guide.
C.3 – C.5		Conforms	
Regulatory Guide 1.37, Rev. 1, 3/07 – Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants			
Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions

Regulatory Guide 1.38, Rev. 2, 5/77 – Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants

Conformance for DCD scope of work is as stated in the DCD. Conformance for remaining scope is documented below.

General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.

Regulatory Guide 1.39, Rev. 2, 9/77 – Housekeeping Requirements for Water-Cooled Nuclear Power Plants

Conformance for DCD scope of work is as stated in the DCD. Conformance for remaining scope is documented below.

General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.

Regulatory Guide 1.45, Rev. 0, 5/73 – Reactor Coolant Pressure Boundary Leakage Detection Systems

Conformance of the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.

C.7		Conforms	

Regulatory Guide 1.52, Rev. 3, 6/01 – 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

Conformance with the design and operational aspects is as stated in the DCD.

Regulatory Guide 1.53, Rev. 2, 11/03 – Application of the Single-Failure Criterion to Safety Systems

Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.

Regulatory Guide 1.54, Rev. 1, 7/00 – Service Level I, II, And III Protective Coatings Applied To Nuclear Power Plants

Conformance of the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.

General		Conforms	

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 1.57, Rev. 1, 3/07 – Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components			
Conformance with Revision 0 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.			
Regulatory Guide 1.59, Rev. 2, 8/77 – Design Basis Floods for Nuclear Power Plants (VEGP COL 1.9-1)			
		Note a	
Regulatory Guide 1.60, Rev. 1, 12/73 – Design Response Spectra for Seismic Design of Nuclear Power Plants			
		Note a	
Regulatory Guide 1.61, Rev. 1, 3/07 – Damping Values for Seismic Design of Nuclear Power Plants			
Conformance with Revision 0 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.			
Regulatory Guide 1.65, Rev. 0, 10/73 – Materials and Inspections for Reactor Vessel Closure Studs			
Conformance of the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
C.3		Conforms	
C.4		Exception	ASME XI ISI criteria for reactor vessel closure stud examinations are applied in lieu of the ASME III NB 2545 and NB 2546 surface examinations. The volumetric examinations currently required by ASME XI provide improved (since 1973) detection of bolting degradation.
Regulatory Guide 1.68, Rev. 3, 3/07 – Initial Test Program for Water-Cooled Nuclear Power Plants			
Conformance with Revision 2 of the Regulatory Guide is documented in the DCD. Conformance of the design aspects is as stated in the DCD. Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C2-C.9 Appendix B Appendix C		Conforms	

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 1.70, Rev. 3, 11/78, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition) (VEGP COL 1.9-1)			
General		Exception	The format and content of the FSAR follow Regulatory Guide 1.206 and AP 1000 Design Control Document as required by Appendix D of 10 CFR Part 52. Note a
Regulatory Guide 1.71, Rev. 1, 3/07 – Welder Qualification for Areas of Limited Accessibility			
Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated in the DCD. Conformance with Revision 1 of the Regulatory Guide during the operational phase (i.e., after the construction phase is completed per the DCD) is documented below.			
General		Conforms	
Regulatory Guide 1.75, Rev. 3, 2/05 – Criteria for Independence of Electrical Safety Systems			
Conformance with Revision 2 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.			
Regulatory Guide 1.76, Rev. 1, 3/07 – Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (VEGP COL 1.9-1)			
Conformance with Revision 0 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.			
		Note a	
Regulatory Guide 1.78, Rev. 1, 12/01 – Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
		Note a	
Regulatory Guide 1.82, Rev. 3, 11/03 – Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
C.1.1.2		Conforms	
C.1.1.5		Conforms	

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions

Regulatory Guide 1.83, Rev. 1, 7/75 - Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes

Conformance of the design aspects is as stated in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (74 FR 58324, 11/12/2009).

Regulatory Guide 1.84, Rev. 33, 8/05 – Design, Fabrication, and Materials Code Case Acceptability, ASME Section III

Conformance with Revision 32 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.

Regulatory Guide 1.86, Rev. 0, 6/74 - Termination of Operating Licenses for Nuclear Reactors

This Regulatory Guide is outside the scope of the FSAR.

VEGP COL 1.9-1

Regulatory Guide 1.91, Rev. 1, 2/78 – Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants

Conformance of the design aspects is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

Note a

Regulatory Guide 1.92, Rev. 2, 07/06 – Combining Modal Responses and Spatial Components in Seismic Response Analysis

Conformance with Revision 1 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.

Regulatory Guide 1.94, Rev. 1, 4/76 – Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

Conformance for DCD scope of work is as stated in the DCD. Conformance for remaining scope is documented below.

Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.

STD COL 1.9-1 (Unless Otherwise Noted)	**Criteria Section**	**Referenced Criteria**	**FSAR Position**	**Clarification/ Summary Description of Exceptions**
	Regulatory Guide 1.97, Rev. 4, 6/06 – Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants			
	Conformance with Revision 3 of the Regulatory Guide is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
	General		Exception	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.
VEGP COL 1.9-1	**Regulatory Guide 1.101, Rev. 5, 6/05 – Emergency Response Planning and Preparedness for Nuclear Power Reactors**			
			Note a	
	Regulatory Guide 1.102, Rev. 1, 9/76 – Flood Protection for Nuclear Power Plants			
			Note a	
	Regulatory Guide 1.109, Rev. 1, 10/77 – 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			
	Conformance of the design aspects is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
			Note a	
	Regulatory Guide 1.110, Rev. 0, 3/76 – Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors			
	Conformance of the design aspects is as stated in the DCD. Conformance with Revision 0 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
	General		Conforms	
VEGP COL 1.9-1	**Regulatory Guide 1.111, Rev. 1, 7/77 – Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors**			
			Note a	

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 1.112, Rev. 1, 3/07 – Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors			
Conformance of the design aspects with Revision 0-R of the Regulatory Guide is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General	ANSI 18.1-1999	Conforms	
		Note a	
Regulatory Guide 1.113, Rev. 1, 4/77 – Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I			
		Note a	
Regulatory Guide 1.114, Rev. 2, 5/89 – Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit			
General		Conforms	
Regulatory Guide 1.115, Rev. 1, 7/77 – Protection Against Low-Trajectory Turbine Missiles			
Conformance of the design aspects is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 1.116, Rev. 0-R, 5/77 – Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems			
Conformance for DCD scope of work is as stated in the DCD. Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Regulatory Guide 1.124, Rev. 2, 02/07 – Service Limits and Loading Combinations for Class 1 Linear-Type Supports			
Conformance with Revision 1 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.			

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
VEGP COL 1.9-1 **Regulatory Guide 1.125, Rev. 1, 10/78 – Physical Models for Design and Operation of Hydraulic Structures and Systems of Nuclear Power Plants**			
		Note a	
Regulatory Guide 1.128, Rev. 2, 2/07 – Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants			
Conformance with Revision 1 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.			
Regulatory Guide 1.129, Rev. 2, 2/07 – Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants			
General	IEEE Std. 450-2002	Exception	Approved Generic Technical Specifications are based on IEEE Std 450-1995.
Regulatory Guide 1.130, Rev. 2, 3/07 - Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Supports			
Conformance with Revision 1 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.			
VEGP COL 1.9-1 **Regulatory Guide 1.132, Rev. 2, 10/03 – Site Investigations for Foundations of Nuclear Power Plants**			
		Note a	
Regulatory Guide 1.133, Rev. 1, 5/81 – Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors			
Conformance of the design aspects is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.2b		Conforms	Procedures are addressed in Section 13.5
C.3a		Conforms	Procedures are addressed in Section 13.5
C.4g		Conforms	Procedures are addressed in Section 13.5
C.4h		Conforms	Procedures are addressed in Section 13.5
C.4i		Conforms	ALARA is addressed in Chapter 12 and Section 13.5
C.4j		Conforms	Training is addressed in Section 13.2

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
C.6		Exception	Regulatory Guide 1.16 has been withdrawn. Event reporting is performed in accordance with 10 CFR 50.72 and 50.73 utilizing the guidance of NUREG-1022
Regulatory Guide 1.134, Rev. 3, 3/98 – Medical Evaluation of Licensed Personnel at Nuclear Power Plants			
General		Conforms	
Regulatory Guide 1.135, Rev. 0, 9/77 – Normal Water Level and Discharge at Nuclear Power Plants			
Conformance of the design aspects is as stated in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (74 FR 39349, 08/06/2009).			
VEGP COL 1.9-1 **Regulatory Guide 1.138, Rev. 2, 12/03 – Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants**			
		Note a	
Regulatory Guide 1.139, Rev. 0, 5/78 – Guidance for Residual Heat Removal			
Conformance with the design aspects is as stated in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (73 FR 32750, 06/10/2008).			
Regulatory Guide 1.143, Rev. 2, 11/01 – Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants			
Conformance of the design aspects is as stated in the DCD. Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
VEGP COL 1.9-1 **Regulatory Guide 1.145, Rev. 1, 11/82 (Revised 2/83 to correct page 1.145-7) – Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants**			
		Note a	
Regulatory Guide 1.147, Rev. 15, 10/07 – Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1			
Conformance with Revision 12 of the Regulatory Guide is documented in the DCD. Conformance of the design aspects is as stated in the DCD. Conformance with Revision 15 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 1.149, Rev. 3, 10/01 – Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations			
C.1		Conforms	During cold licensing, training is conducted using a simulator with limited scope in accordance with Appendix D of ANSI/ANS-3.5-1998. Operator Licensing examinations are conducted on a simulator meeting the applicable requirements of ANSI/ANS-3.5-1998.
Regulatory Guide 1.150, Rev. 1, 2/83 – Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations			
Conformance with the design aspects is as stated in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (73 FR 7766, 02/11/2008).			
Regulatory Guide 1.152, Rev. 2, 1/06 – Criteria for Use of Computers in Safety Systems of Nuclear Power Plants			
Conformance of the design aspects with Revision 1 of the Regulatory Guide is as stated in the DCD. Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Exception	The Cyber Security Program is based on March 2009 revisions of the 10 CFR 73.54 regulations in lieu of Revision 2 of this Regulatory Guide.
Regulatory Guide 1.154, Rev. 0, 1/87 – Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors			
General		Conforms	
Regulatory Guide 1.159, Rev. 1, 10/03 – Assuring the Availability of Funds for Decommissioning Nuclear Reactors			
General		N/A	This Regulatory Guide is outside the scope of the FSAR.
Regulatory Guide 1.160, Rev. 2, 3/97 – Monitoring the Effectiveness of Maintenance at Nuclear Power Plants			
General		Conforms	
Regulatory Guide 1.162, Rev. 0, 2/96 – Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels			
		N/A	This Regulatory Guide is outside the scope of the FSAR.

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 1.163, Rev. 0, 9/95 – Performance-Based Containment Leak-Test Program			
Conformance of the design aspects is as stated in the DCD. Conformance with Revision 0 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
VEGP COL 1.9-1 **Regulatory Guide 1.165, Rev. 0, 3/97 – Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion**			
		Note a	
Regulatory Guide 1.166, Rev. 0, 3/97 – Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions			
General		Conforms	
Regulatory Guide 1.167, Rev. 0, 3/97 – Restart of a Nuclear Power Plant Shut Down by a Seismic Event			
General		Conforms	
Regulatory Guide 1.168, Rev. 1, 2/04 – Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 1.174, Rev. 1, 11/02 – An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis			
This Regulatory Guide is outside the scope of the FSAR.			
Regulatory Guide 1.175, Rev. 0, 8/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing			
Risk-informed inservice testing is not being utilized for this plant.			
Regulatory Guide 1.177, Rev. 0, 8/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications			
General		Conforms	
Regulatory Guide 1.178, Rev. 1, 9/03 – An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping			
Risk-informed inservice inspection is not being utilized for this plant.			

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 1.179, Rev. 0, 1/99 – Standard Format and Content of License Termination Plans for Nuclear Power Reactors			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Regulatory Guide 1.180, Rev. 1, 10/03 – Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems			
Conformance of the design aspects is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	Exclusion zones are established through administrative controls to prohibit the activation of portable EMI/RFI emitters (e.g., welders and transceivers) in areas where safety-related I&C systems are installed.
Regulatory Guide 1.181, Rev. 0, 9/99 – Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)			
General		Conforms	
Regulatory Guide 1.182, Rev. 0, 5/00 – Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants			
General		Conforms	
VEGP COL 1.9-1 **Regulatory Guide 1.183, Rev. 0, 7/00 – Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors**			
		Note a	
Regulatory Guide 1.184, Rev. 0, 7/00 – Decommissioning of Nuclear Power Reactors			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Regulatory Guide 1.185, Rev. 0, 7/00 – Standard Format and Content for Post-shutdown Decommissioning Activities Report			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Regulatory Guide 1.186, Rev. 0, 12/00 – Guidance and Examples for Identifying 10 CFR 50.2 Design Bases			
		N/A	This Regulatory Guide is outside the scope of the FSAR.

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Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 1.187, Rev. 0, 11/00 – Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments			
General		Conforms	
Regulatory Guide 1.188, Rev. 1, 9/05 – Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Regulatory Guide 1.189, Rev. 1, 3/07 – Fire Protection for Nuclear Power Plants			
Conformance with Revision 0 of the Regulatory Guide is documented in the DCD. Conformance of the design aspects is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Regulatory Guide 1.191, Rev. 0, 5/01 – Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Regulatory Guide 1.192, Rev. 0, 6/03 – Operation and Maintenance Code Case Acceptability, ASME OM Code			
General		Conforms	
Regulatory Guide 1.193, Rev. 1, 8/05– ASME Code Cases Not Approved for Use			
General		Conforms	
Regulatory Guide 1.194, Rev. 0, 6/03 – Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants			
General		Conforms	
Regulatory Guide 1.195, Rev. 0, 5/03 – Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors			
This Regulatory Guide is not applicable to the AP1000 certified design.			

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(Unless Otherwise Noted)

Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions

Regulatory Guide 1.196, Rev. 1, 1/07 – Control Room Habitability at Light-Water Nuclear Power Reactors

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below. This Regulatory Guide is not applicable to the AP1000 certified design.

General		Conforms	

Regulatory Guide 1.197, Rev. 0, 5/03 – Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors

Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.

General		Conforms	

VEGP COL 1.9-1

Regulatory Guide 1.198, Rev. 0, 11/03 – Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites

		Note a	

Regulatory Guide 1.199, Rev. 0, 11/03 – Anchoring Components and Structural Supports in Concrete

Conformance with Revision 0 of the Regulatory Guide is as stated in the DCD. This guidance is completely within the scope of the DCD.

Regulatory Guide 1.200, Rev. 1, 1/07 – An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities

General		Conforms	

Regulatory Guide 1.201, Rev. 1, 5/06 – Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance

This Regulatory Guide is not applicable to the AP1000 certified design.

Regulatory Guide 1.202, Rev. 0, 2/05 – Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors

This Regulatory Guide is outside the scope of the FSAR.

Regulatory Guide 1.203, Rev. 0, 12/05 – Transient and Accident Analysis Methods

This Regulatory Guide is not applicable to the AP1000 certified design.

Regulatory Guide 1.204, Rev. 0, 11/05 – Guidelines for Lightning Protection of Nuclear Power Plants

General		Conforms	

STD COL 1.9-1
(Unless Otherwise Noted)

Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 1.205, Rev. 0, 5/06 – Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants			
This Regulatory Guide is not applicable to the AP1000 certified design.			
Regulatory Guide 1.206, Rev. 0, 6/07 – Combined License Applications for Nuclear Power Plants (LWR Edition)			
General	Format	Conforms	
General	Content	Exception	Exceptions to content are identified in Table 1.9-202.
Regulatory Guide 1.207, Rev. 0, 3/07 – Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors			
This Regulatory Guide is not applicable to the AP1000 certified design.			
VEGP COL 1.9-1 **Regulatory Guide 1.208, Rev. 0, 3/07 – A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion**			
		N/A	Performance-based analysis per RG 1.165 and ASCE 43-05. See note a.
Regulatory Guide 1.209, Rev. 0, 3/07 – Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants			
This Regulatory Guide is not applicable to the AP1000 certified design.			
DIVISION 4 — Environmental and Siting			
VEGP COL 1.9-1 **Regulatory Guide 4.2 & Supp. 1, Rev. 2, 7/76, S-1, 9/00 – Preparation of Environmental Reports for Nuclear Power Stations**			
		Note a	
Regulatory Guide 4.7 Rev. 2, 4/98 – General Site Suitability Criteria for Nuclear Power Stations			
		Note a	
Regulatory Guide 4.15 Rev. 2, 7/07 – Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) – Effluent Streams and the Environment			
		Exception	The Guidance of Rev. 1, February 1979 will be followed as per the justification provided in FSAR Subsection 11.5.3.

STD COL 1.9-1
(Unless Otherwise Noted)

Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions

DIVISION 5 — Materials and Plant Protection

The plant-specific physical security plans include no substantive deviations from the NRC-endorsed template in NEI 03-12, Rev. 6. Therefore, the degree of conformance with Division 5 regulatory guides for the Physical Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan is consistent with the degree of conformance of NEI 03-12, Rev. 6.

Regulatory Guide 5.9 Rev. 2, 12/83 – Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material

		N/A	This Regulatory Guide is outside the scope of the FSAR.

Regulatory Guide 5.12, Rev. 0, 11/73 – General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials

Conformance of the design aspects is as stated in the DCD.

		N/A	This Regulatory Guide is outside the scope of the FSAR.

Regulatory Guide 5.65, Rev. 0, 9/86 – Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls

Conformance of the design aspects is as stated in the DCD.

		N/A	This Regulatory Guide is outside the scope of the FSAR.

Regulatory Guide 5.71, Rev. 0, 1/10 – Cyber Security Programs for Nuclear Facilities

Conformance with regulatory positions C.1 through C.5 of Regulatory Guide 5.71, Rev. 0, is as stated in the Cyber Security Plan (CSP), with exceptions to the guidance as noted in Attachment A of the CSP.

DIVISION 8 — Occupational Health

Regulatory Guide 8.2, Rev. 0, 2/73 – Guide for Administrative Practices in Radiation Monitoring

General	10 CFR Part 20; ANSI 13.2-1969	Exception	The reference to 10 CFR 20.401 is no longer valid in the current version of 10 CFR Part 20. ANSI N13.2-1969 was reaffirmed in 1988.

STD COL 1.9-1
(Unless Otherwise Noted)

Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 8.4, Rev. 0, 2/73 - Direct-Reading and Indirect-Reading Pocket Dosimeters			
General	10 CFR Part 20 ANSI N13.5-1972	Exception	The reference to 10 CFR 20.202 (a) and 20.401 is no longer valid in the current version of 10 CFR Part 20. ANSI N13.5-1972 was reaffirmed in 1989. The two performance criteria specified in Regulatory Guide 8.4 (accuracy and leakage) for these devices are met using acceptance standards in ANSI N322-1997 “American National Standard Inspection, Test, Construction, and Performance Requirements for Direct Reading Electrostatic/ Electroscope Type Dosimeters”.
Regulatory Guide 8.5, Rev. 1, 3/81 - Criticality and Other Interior Evacuation Signals			
General		Conforms	
Regulatory Guide 8.6, Rev. 0, 5/73 - Standard Test Procedure for Geiger-Muller Counters			
General		Exception	Instrument calibration program is based upon criteria in ANSI N323A-1997 (with 2004 Correction Sheet) “Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments.” The ANSI 42.3-1969 Standard is no longer recognized as sufficient for calibration of modern instruments.
Regulatory Guide 8.7, Rev. 2, 11/05 - Instructions for Recording and Reporting Occupational Radiation Dose Data			
General		Conforms	

STD COL 1.9-1
(Unless Otherwise Noted)

Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 8.8, Rev. 3, 6/78 – Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable			
Conformance of the design aspects is as stated in the DCD. Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.1		Conforms	
C.3.a		Conforms	
C.3.b		Exception	Regulatory Guide 1.16 C.1.b.(3) data is no longer reported. Reporting per C.1.b(2) is also no longer required.
C.3.c		Conforms	
C.4.b-C.4.d	ANSI Z-88.2, Regulatory Guide 8.15, NUREG-0041	Conforms	Conformance is with the latest revision of NUREG-0041.
Regulatory Guide 8.9, Rev. 1, 7/93 – Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program			
General		Conforms	
Regulatory Guide 8.10, Rev. 1-R, 5/77 – Operating Philosophy For Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable			
General		Conforms	
Regulatory Guide 8.13, Rev. 3, 6/99 – Instruction Concerning Prenatal Radiation Exposure			
General		Conforms	
Regulatory Guide 8.15, Rev. 1, 10/99 – Acceptable Programs for Respiratory Protection			
General		Conforms	
Regulatory Guide 8.27, Rev. 0, 3/81 – Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants			
General		Conforms	
Regulatory Guide 8.28, Rev. 0, 8/81 – Audible-Alarm Dosimeters			
General	ANSI N13.27-1981	Conforms	

STD COL 1.9-1
(Unless Otherwise Noted)

Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
Regulatory Guide 8.29, Rev. 1, 2/96 – Instruction Concerning Risks from Occupational Radiation Exposure			
General		Conforms	
Regulatory Guide 8.34, Rev. 0, 7/92 – Monitoring Criteria and Methods To Calculate Occupational Radiation Doses			
General		Conforms	
Regulatory Guide 8.35, Rev. 0, 6/92 – Planned Special Exposures			
General		Conforms	
Regulatory Guide 8.36, Rev. 0, 7/92 – Radiation Dose to the Embryo/Fetus			
General		Conforms	
Regulatory Guide 8.38, Rev. 1, 5/06 – Control of Access to High and Very High Radiation Areas in Nuclear Power Plants			
Conformance of the design aspects is as stated in the DCD. Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	

Note a. Refer to ESPA SSAR Tables 1-2 and 1-3.

Note b. Above stated general alternatives regarding the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD is provided to preserve the finality of the certified design. Further, each stated conformance with the programmatic and/or operational aspects is only to the extent that a design change or departure from the approved DCD is not required to implement those programmatic and/or operational aspects. As the operational and programmatic aspects become more fully defined (for example, during the preparation, approval, or initial implementation of plant procedures), there exists a potential that a conflict could be identified between the design as certified in the DCD and the programmatic and/or operational aspects of the guidance. In such cases, the design certification (rule) becomes the controlling factor, and the design conformance to the Regulatory Guide is per the revision stated in the DCD.

Note c. 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 operational aspects.”

APPENDIX 1B
SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVES

STD SUP 1B-1 DCD Appendix 1B is not incorporated into this FSAR. Rather, the severe accident mitigation design alternatives are addressed in the Environmental Report. As indicated in 10 CFR Part 52, Appendix D, Section III.B, "...the evaluation of severe accident mitigation design alternatives in appendix 1B of the generic DCD are not part of this appendix."

VEGP SUP 1B-2 The applicable Environmental Report was provided in the Early Site Permit (ESP) application and evaluated in the Environmental Impact Statement for an ESP at the Vogtle Electric Generating Plant Site.

CHAPTER 2

SITE CHARACTERISTICS

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CHAPTER 2
SITE CHARACTERISTICS

The introductory information at the beginning of Chapter 2 of the referenced DCD is incorporated by reference with the following departures and/or supplements.

ESPA SSAR Section 1.3 is incorporated by reference with the following variances and/or supplements.

Insert the following subsection at the end of the introductory text of DCD Chapter 2, prior to DCD Section 2.1.

2.0 SITE CHARACTERISTICS

VEGP SUP 2.0-1 Chapter 2 describes the characteristics and site-related design parameters of the Vogtle Electric Generating Plant (VEGP), Units 3 and 4. The site location, characteristics and parameters, as described in the following five sections are provided in sufficient detail to support a safety assessment:

- Geography and Demography (Section 2.1)
- Nearby industrial, Transportation, and Military Facilities (Section 2.2)
- Meteorology (Section 2.3)
- Hydrologic Engineering (Section 2.4)
- Geology, Seismology, and Geotechnical Engineering (Section 2.5)

Table 2.0-201 provides a comparison of site-related design parameters for which the AP1000 plant is designed and site characteristics specific to VEGP in support of this safety assessment. The first two columns of Table 2.0-201 are a compilation of the site parameters from DCD Table 2-1 and DCD Tier 1 Table 5.0-1. The third column of Table 2.0-201 is the corresponding site characteristic for the VEGP. The fourth column denotes the place where this data is presented. The last column indicates whether or not the site characteristic falls within the AP1000 site parameters. “Yes” indicates the site characteristic falls within the parameter. Control room atmospheric dispersion factors (χ/Q) for accident dose analysis are presented in Table 2.0-202. All of the control room χ/Q values fall within the AP1000 parameters.

VEGP SUP 2.0-1

Table 2.0-201 (Sheet 1 of 9)
Comparison of AP1000 DCD Site Parameters and Vogtle Electric Generating Plant Units 3 & 4 Site Characteristics

		AP1000 DCD Site Parameter[a]	VEGP Site Characteristic	VEGP Reference	VEGP Within Site Parameter
	Air Temperature				
	Maximum Safety[b]	115°F dry bulb/86.1°F coincident wet bulb[h]	115°F dry bulb/77.7°F coincident wet bulb	ESPA SSAR Table 1-1	Yes
		86.1°F wet bulb (noncoincident)	83.9°F wet bulb (noncoincident)	ESPA SSAR Table 1-1	Yes
	Minimum Safety[b]	-40°F	-8°F	ESPA SSAR Table 1-1	Yes
VEGP ESP VAR 2.3-1	Maximum Normal[c]	101°F dry bulb/80.1°F coincident wet bulb	97°F dry bulb/76°F coincident wet bulb	Subsection 2.3.1.5	Yes
		80.1°F wet bulb (noncoincident)[d]	79°F wet bulb (noncoincident)	Subsection 2.3.1.5	Yes
VEGP ESP VAR 2.3-1	Minimum Normal[c]	-10°F	21°F dry bulb	Subsection 2.3.1.5	Yes
	Wind Speed				
	Operating Basis	145 mph (3 second gust); importance factor 1.15 (safety), 1.0 (nonsafety); exposure C; topographic factor 1.0	104 mph (3 second gust); exposure C; topographic factor 1.0. (Importance factor is not a property of the wind speed.)	ESPA SSAR Table 1-1 ESPA SSAR Figure 2.5.1-32	Yes
	Tornado	300 mph	300 mph	ESPA SSAR Table 1-1	Yes
		Maximum pressure differential of 2.0 lb/in^2	2.0 lb/in^2	ESPA SSAR Table 1-1	Yes

VEGP SUP 2.0-1

Table 2.0-201 (Sheet 2 of 9)
Comparison of AP1000 DCD Site Parameters and Vogtle Electric Generating Plant Units 3 & 4 Site Characteristics

	AP1000 DCD Site Parameter[a]	VEGP Site Characteristic	VEGP Reference	VEGP Within Site Parameter
Seismic				
CSDRS	CSDRS free field peak ground acceleration of 0.30 g with modified Regulatory Guide 1.60 response spectra (See Figures 5.0-1 and 5.0-2.). The SSE is now referred to as CSDRS. Seismic input is defined at finished grade except for sites where the nuclear island is founded on hard rock. If the site-specific spectra exceed the response spectra in Figures 5.0-1 and 5.0-2 at any frequency, or if soil conditions are outside the range evaluated for AP1000 design certification, a site-specific evaluation can be performed. This evaluation will consist of a site-specific dynamic analysis and generation of in-structure response spectra at key locations to be compared with the floor response spectra of the certified design at 5-percent damping. The site is acceptable if the floor response spectra from the site-specific evaluation do not exceed the AP1000 spectra for each of the locations or the exceedances are justified.	Site-specific GMRS values specified and illustrated in ESPA SSAR Section 2.5.2. The seismic design of AP-1000 nuclear island is discussed in Section 3.7.1.1.1. Site-specific evaluation performed in ESPA SSAR Appendix 2.5E	ESPA SSAR Table 1-1 FSAR 3.7.1.1.1 ESPA SSAR Appendix 2.5E	Yes

VEGP SUP 2.0-1

Table 2.0-201 (Sheet 3 of 9)
Comparison of AP1000 DCD Site Parameters and Vogtle Electric Generating Plant Units 3 & 4 Site Characteristics

	AP1000 DCD Site Parameter[a]	VEGP Site Characteristic	VEGP Reference	VEGP Within Site Parameter
	The hard rock high frequency (HRHF) envelope response spectra are shown in Figure 5.0-3 and Figure 5.0-4 defined at the foundation level for 5% damping. The HRHF envelope response spectra provide an alternative set of spectra for evaluation of site specific GMRS. A site is acceptable if its site specific GMRS fall within the AP1000 HRHF envelope response spectra. Evaluation of a site for application of the HRHF envelope response spectra includes consideration of the limitation on shear wave velocity identified for use of the HRHF envelope response spectra. This limitation is defined by a shear wave velocity at the bottom of the basemat equal to or higher than 7,500 fps, while maintaining a shear wave velocity equal to or above 8,000 fps at the lower depths.			
Fault Displacement Potential	No potential fault displacement considered beneath the seismic Category I and seismic Category II structures and immediate surrounding area. The immediate surrounding area includes the effective soil supporting media associated with the seismic Category I and seismic Category II structures.	No fault displacement potential within the investigative area.	ESPA SSAR Table 1-1	Yes

Table 2.0-201 (Sheet 4 of 9)
VEGP SUP 2.0-1
Comparison of AP1000 DCD Site Parameters and Vogtle Electric Generating Plant Units 3 & 4 Site Characteristics

	AP1000 DCD Site Parameter[(a)]	VEGP Site Characteristic	VEGP Reference	VEGP Within Site Parameter
Soil				
Average Allowable Static Bearing Capacity	The allowable bearing capacity, including a factor of safety appropriate for the design load combination, shall be greater than or equal to the average bearing demand of 8,900 lb/ft² over the footprint of the nuclear island at its excavation depth	34,000 lb/ft^2	ESPA SSAR Table 1-1	Yes
Dynamic Bearing Capacity for Normal Plus Safe Shutdown Earthquake (SSE)	The allowable bearing capacity, including a factor of safety appropriate for the design load combination, shall be greater than or equal to the maximum bearing demand of 35,000 lb/ft² at the edge of the nuclear island at its excavation depth, or Site-specific analyses demonstrate factor of safety appropriate for normal plus safe shutdown earthquake loads.	42,000 lb/ft^2	ESPA SSAR Table 1-1	Yes
Shear Wave Velocity	Greater than or equal to 1,000 ft/sec based on minimum low-strain soil properties over the footprint of the nuclear island at its excavation depth	Greater than 1000 ft/sec	ESPA SSAR Table 1-1	Yes
Lateral Variability	Soils supporting the nuclear island should not have extreme variations in subgrade stiffness. This may demonstrated by one of the following:			

VEGP SUP 2.0-1

Table 2.0-201 (Sheet 5 of 9)
Comparison of AP1000 DCD Site Parameters and Vogtle Electric Generating Plant Units 3 & 4 Site Characteristics

		AP1000 DCD Site Parameter[a]	VEGP Site Characteristic	VEGP Reference	VEGP Within Site Parameter
Lateral Variability (Continued)	1	Soils supporting the nuclear island are uniform in accordance with Regulatory Guide 1.132 if the geologic and stratigraphic features at depths less than 120 feet below grade can be correlated from one boring or sounding location to the next with relatively smooth variations in thickness or properties of the geologic units, or	Site is uniform based on boring data and placement of engineered backfill	ESPA SSAR 2.5.4.4 and 2.5.4.5	Yes
	2	Site specific assessment of subsurface conditions demonstrates that the bearing pressures below the footprint of the nuclear island do not exceed 120% of those from the generic analyses of the nuclear island at a uniform site, or	N/A		
	3	Site specific analysis of the nuclear island basemat demonstrates that the site specific demand is within the capacity of the basemat.	N/A		
		As an example of sites that are considered uniform, the variation of shear wave velocity in the material below the foundation to a depth of 120 feet below finished grade within the nuclear island footprint and 40 feet beyond the boundaries of the nuclear island footprint meets the criteria in the case outlined below:			

VEGP SUP 2.0-1

Table 2.0-201 (Sheet 6 of 9)
Comparison of AP1000 DCD Site Parameters and Vogtle Electric Generating Plant Units 3 & 4 Site Characteristics

	AP1000 DCD Site Parameter[a]		VEGP Site Characteristic	VEGP Reference	VEGP Within Site Parameter
Lateral Variability (Continued)	Case 1: For a layer with a low strain shear wave velocity greater than or equal to 2500 feet per second, the layer should have approximately uniform thickness, should have a dip not greater than 20 degrees, and should have less than 20 percent variation in the shear wave velocity from the average velocity than any layer.		N/A		
Limits of Acceptable Settlement Without Additional Evaluation[i]	Differential Across Nuclear Island Foundation Mat	1/2 inch in 50 ft	~1/4 inch in 50 ft (projected)	ESPA SSAR 2.5.4.10.2	Yes (projected)
	Total for Nuclear Island Foundation Mat	6 inches	2–3 inches (projected)		
	Differential Between Nuclear Island and Turbine Building[j]	3 inches	<1 inch (projected)		
	Differential Between Nuclear Island and Other Buildings[j]	3 inches	<1 inch (projected)		

VEGP SUP 2.0-1

Table 2.0-201 (Sheet 7 of 9)
Comparison of AP1000 DCD Site Parameters and Vogtle Electric Generating Plant Units 3 & 4 Site Characteristics

	AP1000 DCD Site Parameter[a]	VEGP Site Characteristic	VEGP Reference	VEGP Within Site Parameter
Liquefaction Potential	No liquefaction considered beneath the seismic Category I and seismic Category II structures and immediate surrounding area. The immediate surrounding area includes the effective soil supporting media associated with the seismic Category I and seismic Category II structures.	None at the site-specific SSE.	ESPA SSAR Table 1-1	Yes
Minimum Soil Angle of Internal Friction	Minimum soil angle of internal friction is greater than or equal to 35 degrees below the footprint of nuclear island at its excavation depth. If the minimum soil angle of internal friction is below 35 degrees, a site specific analysis shall be performed using the site specific soil properties to demonstrate stability.	36 degrees	ESPA SSAR Table 1-1	Yes
Missiles				
Tornado	4000-lb automobile at 105 mph horizontal, 74 mph vertical 275-lb, 8-in. shell at 105 mph horizontal, 74 mph vertical 1-inch-diameter steel ball at 105 mph in the most damaging direction	4000-lb automobile at 105 mph horizontal, 74 mph vertical 275-lb, 8-in. shell at 105 mph horizontal, 74 mph vertical 1-inch-diameter steel ball at 105 mph in the most damaging direction	Subsection 3.5.1.5 DCD Section 3.5.1.4 APP-GW-GLR-020, “Wind and Tornado Site Interface Criteria,” Westinghouse Electric Company LLC.[e]	Yes
Flood Level	Less than plant elevation 100 feet	The design basis river flood level is El. 178.10 ft MSL, which is 41.9 feet below plant elevation (220 ft MSL). Maximum local PMP flood elevation is 219.47 ft MSL, which is 0.53 feet below plant elevation (220 ft MSL).	ESPA SSAR Table 1-1 Subsection 2.4.2	Yes

VEGP SUP 2.0-1

Table 2.0-201 (Sheet 8 of 9)
Comparison of AP1000 DCD Site Parameters and Vogtle Electric Generating Plant Units 3 & 4 Site Characteristics

	AP1000 DCD Site Parameter[a]	VEGP Site Characteristic	VEGP Reference	VEGP Within Site Parameter
Ground Water Level	Less than plant elevation 98 feet	The maximum groundwater level is 165 ft MSL which is 55 feet below plant elevation (220 ft MSL).	ESPA SSAR Table 1-1	Yes
Plant Grade Elevation	Less than plant elevation 100 feet, except for portion at a higher elevation adjacent to the annex building	The standard plant-floor elevation of the safety-related facilities is established at plant elevation 220 ft MSL; the finished plant grade elevation slopes away from plant structures	Figure 2.4-201	Yes
Precipitation				
Rain	20.7 in/hr [1-hr 1-mi^2 PMP]	19.2 in/hr	ESPA SSAR Table 1-1	Yes
Snow/Ice	75 pounds per square foot on ground with exposure factor of 1.0 and importance factors of 1.2 (safety) and 1.0 (non-safety)	10.0 pounds per square foot	ESPA SSAR Table 1-1	Yes
Atmospheric Dispersion Values - χ/Q[f]				
Site Boundary (annual average)	$\leq 2.0 \times 10^{-5}$ sec/m^3	0.55×10^{-5} sec/m^3	ESPA SSAR Table 1-1	Yes
VEGP ESP PC 9 Site Boundary (0-2 hr)	$\leq 5.1 \times 10^{-4}$ sec/m^3 [g]	3.49×10^{-4} sec/m^3	ESPA SSAR Table 1-1	Yes
Low population zone boundary[g]				
0–8 hr	$\leq 2.2 \times 10^{-4}$ sec/m^3	7.04×10^{-5} sec/m^3	ESPA SSAR Table 1-1	Yes
8–24 hr	$\leq 1.6 \times 10^{-4}$ sec/m^3	5.25×10^{-5}sec/m^3	ESPA SSAR Table 1-1	Yes
24–96 hr	$\leq 1.0 \times 10^{-4}$ sec/m^3	2.77×10^{-5} sec/m^3	ESPA SSAR Table 1-1	Yes
96–720 hr	$\leq 8.0 \times 10^{-5}$ sec/m^3	1.11×10^{-5} sec/m^3	ESPA SSAR Table 1-1	Yes
Control Room	Table 2.0-202	Table 2.0-202	Table 2.0-202	Yes

Table 2.0-201 (Sheet 9 of 9)

VEGP SUP 2.0-1

Comparison of AP1000 DCD Site Parameters and Vogtle Electric Generating Plant Units 3 & 4 Site Characteristics

	AP1000 DCD Site Parameter[a]	VEGP Site Characteristic	VEGP Reference	VEGP Within Site Parameter
Population Distribution[g]				
Exclusion area (site)	0.5 mi.	The minimum distance from the effluent release boundary to the exclusion area boundary is 0.50 mile.[f]	ESPA SSAR Table 1-1	Yes

a) AP1000 DCD Site Parameters are a compilation of DCD Tier 1 Table 5.0-1 and DCD Tier 2 Table 2-1.
b) Maximum and minimum safety values are based on historical data and exclude peaks of less than 2 hours duration.
c) The maximum normal value is the 1-percent seasonal exceedance 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 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 99-percent seasonal exceedance is approximately equivalent to the annual 99.6-percent exceedance. See Subsection 2.3.1.5 for further discussion on this relationship.
d) The noncoincident wet bulb temperature is applicable to the cooling tower only.
e) Per APP-GW-GLR-020, the kinetic energies of the missiles discussed in DCD Section 3.5 are greater than the kinetic energies of the missiles discussed in Regulatory Guide 1.76 and result in a more conservative design.
f) For AP1000, the term “site boundary” and “exclusion area boundary” are used interchangeably. Thus, the χ/Q specified for the site boundary applies whenever a discussion refers to the exclusion area boundary. At VEGP the “site boundary” and ”exclusion area boundary” are not interchangeable. See Figure 1.1-202.
g) Site Interface Values for Post-Accident Dose Consequences and Minimum Distance to Site Boundary are reported per ESPA SSAR Section 1.3 and Table 1-1. Cooling Tower Make-up Flow Rate, which is not an AP1000 DCD Site Parameter, is 61,145 gpm (2 units) per ESPA SSAR Table 1-1.
h) The containment pressure response analysis is based on a conservative set of dry-bulb and wet-bulb temperatures. These results envelope any conditions where the dry-bulb temperature is 115°F or less and wet-bulb temperature is less than or equal to 86.1°F.
i) Additional evaluation may include evaluation of the impact of the elevated estimated settlement values on the critical components of the AP1000, determining a construction sequence to control the predicted settlement behavior, or developing an active settlement monitoring system throughout the entire construction sequence as well as a long-term (plant operation) plan.
j) Differential settlement is measured at center of Nuclear Island and center of adjacent structures.

Table 2.0-202
Comparison of Control Room Atmospheric Dispersion Factors for Accident Analysis for AP1000 DCD and VEGP Units 3 & 4 (Sheet 1 of 2)

VEGP SUP 2.0-1

VEGP ESP PC 9

X/Q (sec/m^3) at HVAC Intake for the Identified Release Points[a]

	Plant Vent or PCS Air Diffuser[b]	Plant Vent	PCS Air Diffuser	**Ground Level Containment Release Points[c]**	Ground Level Containment Release Points	**PORV and Safety Valve Releases[d]**	PORV and Safety Valve Releases	**Condenser Air Removal Stack[g]**	Condenser Air Removal Stack	**Steam Line Break Releases**	Steam Line Break Releases	**Fuel Handling Area[e]**	Fuel Handling Area Blowout Panel	Fuel Handling Area Truck Bay Door
Release Time	**DCD**	VEGP	VEGP	**DCD**	VEGP	**DCD**	VEGP	**DCD**	VEGP	**DCD**	VEGP	**DCD**	VEGP	VEGP
0 – 2 hours	**3.0E-3**	2.02E-03	1.68E-03	**6.0E-3**	3.20E-03	**2.0E-2**	1.31E-02	**6.0E-3**	1.54E-03	**2.4E-2**	1.48E-02	**6.0E-3**	1.54E-03	1.15E-03
2 – 8 hours	**2.5E-3**	1.58E-03	1.29E-03	**3.6E-3**	1.82E-03	**1.8E-2**	1.02E-02	**4.0E-3**	1.17E-03	**2.0E-2**	1.20E-02	**4.0E-3**	1.11E-03	8.29E-04
8 – 24 hours	**1.0E-3**	6.37E-04	5.47E-04	**1.4E-3**	8.27E-04	**7.0E-3**	4.62E-03	**2.0E-3**	5.36E-04	**7.5E-3**	5.41E-03	**2.0E-3**	4.42E-04	3.35E-04
1 – 4 days	**8.0E-4**	5.12E-04	4.55E-04	**1.8E-3**	7.22E-04	**5.0E-3**	3.29E-03	**1.5E-3**	3.94E-04	**5.5E-3**	3.93E-03	**1.5E-3**	3.57E-04	2.62E-04
4 – 30 days	**6.0E-4**	3.82E-04	3.34E-04	**1.5E-3**	5.70E-04	**4.5E-3**	2.77E-03	**1.0E-3**	2.78E-04	**5.0E-3**	3.26E-03	**1.0E-3**	2.59E-04	1.86E-04

X/Q (sec/m^3) at Annex Building Door for the Identified Release Points[f]

	Plant Vent or PCS Air Diffuser[b]	Plant Vent	PCS Air Diffuser	**Ground Level Containment Release Points[c]**	Ground Level Containment Release Points	**PORV and Safety Valve Releases[d]**	PORV and Safety Valve Releases	**Condenser Air Removal Stack[g]**	Condenser Air Removal Stack	**Steam Line Break Releases**	Steam Line Break Releases	**Fuel Handling Area[e]**	Fuel Handling Area Blowout Panel	Fuel Handling Area Truck Bay Door
Release Time	**DCD**	VEGP	VEGP	**DCD**	VEGP	**DCD**	VEGP	**DCD**	VEGP	**DCD**	VEGP	**DCD**	VEGP	VEGP
0 – 2 hours	**1.0E-3**	4.32E-04	4.48E-04	**1.0E-3**	3.93E-04	**4.0E-3**	9.81E-04	**2.0E-2**	4.00E-03	**4.0E-3**	9.23E-04	**6.0E-3**	3.77E-04	3.48E-04
2 – 8 hours	**7.5E-4**	3.52E-04	3.38E-04	**7.5E-4**	3.16E-04	**3.2E-3**	7.69E-04	**1.8E-2**	3.15E-03	**3.2E-3**	7.31E-04	**4.0E-3**	2.84E-04	2.60E-04
8 – 24 hours	**3.5E-4**	1.44E-04	1.44E-04	**3.5E-4**	1.32E-04	**1.2E-3**	3.12E-04	**7.0E-3**	1.35E-03	**1.2E-3**	2.98E-04	**2.0E-3**	1.18E-04	1.09E-04
1 – 4 days	**2.8E-4**	1.15E-04	1.17E-04	**2.8E-4**	1.07E-04	**1.0E-3**	2.49E-04	**5.0E-3**	1.04E-03	**1.0E-3**	2.37E-04	**1.5E-3**	9.50E-05	8.75E-05
4 – 30 days	**2.5E-4**	8.47E-05	8.77E-05	**2.5E-4**	8.14E-05	**8.0E-4**	1.87E-04	**4.5E-3**	8.05E-04	**8.0E-4**	1.75E-04	**1.0E-3**	6.83E-05	6.16E-05

Table 2.0-202
Comparison of Control Room Atmospheric Dispersion Factors for Accident Analysis for AP1000 DCD and VEGP Units 3 & 4 (Sheet 2 of 2)

VEGP SUP 2.0-1

a. These dispersion factors are to be used 1) for the time period preceding the isolation of the main control room and actuation of the emergency habitability system, 2) for the time after 72 hours when the compressed air supply in the emergency habitability system would be exhausted and outside air would be drawn into the main control room, and 3) for the determination of control room doses when the non-safety ventilation system is assumed to remain operable such that the emergency habitability system is not actuated.

b. These dispersion factors are used for analysis of the doses due to a postulated small line break outside of containment. The plant vent and PCS air diffuser are potential release paths for other postulated events (loss of-coolant accident, rod ejection accident, and fuel handling accident inside the containment); however, the values are bounded by the dispersion factors for ground level releases.

c. The listed values represent modeling the containment shell as a diffuse area source, and are used for evaluating the doses in the main control room for a loss-of-coolant accident, for the containment leakage of activity following a rod ejection accident, and for a fuel handling accident occurring inside the containment.

d. The listed values bound the dispersion factors for releases from the steam line safety & power-operated relief valves. These dispersion factors would be used for evaluating the doses in the main control room for a steam generator tube rupture, a main steam line break, a locked reactor coolant pump rotor, and for the secondary side release from a rod ejection accident.

e. The listed values bound the dispersion factors for releases from the fuel storage and handling area. The listed values also bound the dispersion factors for releases from the fuel storage area in the event that spent fuel boiling occurs and the fuel handling area relief panel opens on high temperature. These dispersion factors are used for the fuel handling accident occurring outside containment and for evaluating the impact of releases associated with spent fuel pool boiling.

f. These dispersion factors are to be used when the emergency habitability system is in operation and the only path for outside air to enter the main control room is that due to ingress/egress.

g. This release point is included for information only as a potential activity release point. None of the design basis accident radiological consequences analyses model release from this point.

2.1 GEOGRAPHY AND DEMOGRAPHY

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

This section of the referenced ESPA SSAR is incorporated by reference with no variances or supplements.

VEGP DEP 1.1-1 Subsection 2.1.1 of the DCD is renumbered as Subsection 2.1.4 and moved to the end of Section 2.1. This is being done to accommodate the incorporation of Regulatory Guide 1.206 numbering conventions for Section 2.1.

VEGP DEP 1.1-1

2.1.4 COMBINED LICENSE INFORMATION FOR GEOGRAPHY AND DEMOGRAPHY

VEGP COL 2.1-1 This COL item is addressed in Subsections 1.1.1 and 1.2.2 and in ESPA SSAR Section 2.1.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

This section of the referenced DCD is incorporated by reference with the following departure(s) and/or supplement(s).

This section of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

VEGP DEP 1.1-1 Subsection 2.2.1 of the DCD is renumbered as Subsection 2.2.4 and moved to the end of Section 2.2. This is being done to accommodate the incorporation of Regulatory Guide 1.206 numbering conventions for Section 2.2.

2.2.3 EVALUATION OF POTENTIAL ACCIDENTS

2.2.3.2 Hazardous Chemicals

Add the following after the first paragraph in ESPA SSAR Section 2.2.3.2.3.

The impact on the new Units 3 and 4 due to an accidental hydrazine release is evaluated in Subsection 2.2.3.2.3.1 below.

VEGP COL 2.2-1
VEGP COL 6.4-1
VEGP ESP COL 2.2-1

2.2.3.2.3.1 Hydrazine Hazard from Onsite Storage Tanks

Impact on safety related structures and control room habitability for Units 3 and 4 due to accidental releases from or explosion in the 6,644 gallon Units 1 and 2 hydrazine tank was not evaluated in the ESPA SSAR, but is evaluated below.

The Areal Locations of Hazardous Atmospheres (ALOHA) code (Reference 202) and the TNT equivalency method is used to determine the minimum safe distances for hydrazine that is stored onsite at VEGP. These minimum safe distances for Unit 3 control room are then compared to the distances from where hydrazine is stored to Unit 3. Since the Unit 4 control room is further west of Unit 3, the evaluation is based on Unit 3 only and then the results are applied to Unit 4. The four scenarios evaluated are: toxicity of a vapor cloud, flammability of a vapor cloud, explosive vapor cloud, and a tank explosion.

The assumptions for the three vapor cloud scenarios include the following:

- Hydrazine is a 35% hydrazine solution.

- Atmospheric air flow is turbulent in only one direction (no cross flow) such that the released gases spread downstream in a Gaussian manner.

- Total quantity of hydrazine is released and forms an evaporating puddle with a depth of 1 cm (NUREG-0570). This provides a significant surface area to maximize evaporation and the formation of a vapor cloud.

- Ambient temperature is 95.1°F for daytime releases and 70.1°F for nighttime releases, the relative humidity is 50%, and the atmospheric pressure is 1 atmosphere (40 CFR 68.22).

- A sensitivity study was performed to determine the worst-case meteorological conditions (wind speed and stability class). The worst-case scenario is a wind speed of 2 m/s and stability class “F”.

- Ground roughness is “Urban or Forest” which most accurately represents site conditions.

- Cloud cover selected is based on the appropriate stability class and wind speed (Reference 202).

- Time of accidental release is 12:00 pm on July 21, 2008 for daytime releases and 5:00 a.m. on July 21, 2008 for nighttime releases. The date was selected because it coincides with the highest daily maximum temperature, and 12:00 p.m. was selected because solar radiation is highest during midday. Higher solar radiation leads to a higher evaporation rate and thus a larger vapor cloud. Five o’clock (5:00) a.m. on July 21, 2008 was selected to provide a realistic meteorological condition for the more stable stability classes. ALOHA requires manual override if 12:00 p.m. is used with stability classes “E” and “F”, or “D” with a wind speed of 3 m/s (Reference 202).

- Wind input height is 10 meters. ALOHA calculates a wind profile based on where the meteorological data is taken. ALOHA assumes that the MET station is at a height of 10 meters. The National Weather Service usually reports wind speeds from the height of 10 meters.

- There is no temperature inversion.

- It is not known how long after a release ignition occurs for vapor cloud explosions. Therefore, the “unknown” time of vapor cloud ignition option was selected for this case. ALOHA will run explosion scenarios for a range of ignition times that encompass all of the possible ignition times for a scenario. ALOHA takes the results from all of these scenarios and combines them on a single Threat Zone plot.

- Type of vapor cloud ignition is “ignited by detonation.” This is the worst case scenario for an accidental explosion.

The assumptions for the tank explosion (TNT mass equivalency) scenario include the following:

- Vapor space is assumed to be the tank volume at the upper flammability limit of hydrazine.
- Air temperature is 32.2°F, the lowest mean daily minimum temperature, which corresponds to an air density of 0.081 lb/ft3.
- Detonation occurs inside the tank.
- Vapor explosion is treated as if it is completely confined. Thus, a yield factor of 100% is used for the confined vapor explosion (NUREG-1805).

Toxicity of a Hydrazine Vapor Cloud

For assessing the toxicity of a vapor cloud from hydrazine release, it is necessary to determine the maximum distance at which the Immediately Dangerous to Life or Health (IDLH) value exists (Regulatory Guide 1.78). This distance represents the minimum safe distance from the hydrazine storage area that a nuclear power plant can operate. The distance depends on the prevailing meteorological conditions, wind speed, relative humidity, atmospheric pressure, ambient temperature, toxicity and the quantity of hydrazine released. It is also necessary to determine the resulting concentration of hydrazine inside the control room to ascertain the effects of a toxic vapor on the operators. ALOHA calculated both the inside and outside concentrations of the control room over time (0 to 1 hour). For this evaluation, a release of 6,644 gallons of 35% hydrazine solution is assumed.

The hydrazine tank is located east of the Unit 1 Turbine Building, 2,200 feet from the Unit 3 control room. The evaluation considers a control room air exchange rate of 0.95 exchanges per hour, and an IDLH for hydrazine of 50 ppm. The maximum vapor cloud distance to the IDLH is calculated to be 927 feet (the resulting maximum concentration at the control room air intake is 15.4 ppm). The maximum concentration of hydrazine inside the control room is calculated to be 7.76 ppm. The resulting hydrazine concentrations inside the Units 3 and 4 control rooms are within the IDLH limit value of 50 ppm.

Results indicate that operators in the Units 3 and 4 control rooms are not impacted by the potential toxicity from a hydrazine vapor cloud.

Flammability of a Hydrazine Vapor Cloud

For assessing the flammability of a vapor cloud from a hydrazine release, the ALOHA air dispersion model is used to determine the distances where the vapor cloud may exist between the upper flammability limit (UFL) and lower flammability limit (LFL) (40 CFR 68.22). Once the concentration of the hydrazine vapor cloud is above the UFL or below the LFL, the vapor is no longer flammable.

For this evaluation, a release of 6,644 gallons of 35% hydrazine solution is analyzed for potential flammable hydrazine vapor threats.

Hydrazine has an LFL of 4.7% and a UFL of 99.9%. The distance from the leak source to the LFL is 54 feet. Though ALOHA does report a distance to the LFL, the vapor cloud does not ever exceed the LFL for any scenario. The distance that is reported is the same for every situation due to near field patchiness. It is further shown that the LFL is never exceeded because, as shown below, no explosions occur, even though a detonation was chosen in every instance.

The distance from the hydrazine storage tank to where the hydrazine vapor cloud exists between the UFL and the LFL is less than the distance from the storage tank to the Units 3 and 4 control rooms. Therefore, results indicate that there is no potential flammable, hydrazine vapor cloud reaching safety related structures or the operators in the Units 3 and 4 control rooms.

Explosive Hydrazine Vapor Cloud

For assessing the explosion from a vapor cloud due to hydrazine release, it is necessary to determine the “safe distance”, the minimum distance required for an explosion to have less than or equal to 1 psi peak incident pressure (Regulatory Guide 1.91). This is the minimum safe distance for no impacts from an explosion of a hydrazine vapor cloud. A peak overpressure of 1 psi will shatter glass but not significantly cause structural damage to buildings (Regulatory Guide 1.91). The peak overpressure to the Unit 3 control room is also established. For this evaluation, a release of 6,644 gallons of 35% hydrazine solution is analyzed for potential explosive vapor threats.

The ALOHA analysis indicates that the vapor cloud does not reach the LFL and, therefore, does not explode. Since there is no explosion, the safety related structures and operators working in the Units 3 and 4 control rooms are not impacted.

Hazard from a Tank Explosion

The methodology presented below is for a confined explosion occurring within some form of a storage container (i.e. tank). Since only vapor will burn or explode, the methodology employed considers the maximum vapor within the hydrazine storage tank as explosive (equivalent TNT). For atmospheric liquid storage, this maximum vapor would involve the container to be completely empty of liquid and filled only with air and chemical vapor at UFL conditions (NUREG-1805). Due to complete confinement and the use of only the UFL vapor mass, a 100% yield factor is attributed to the explosion (NUREG-1805). The equivalent mass of TNT is calculated by taking into account the product of the vapor mass (within the flammable range), heat of combustion, and the explosion yield factor. Once the equivalent mass of TNT is calculated, a radial distance generating a 1 psi peak incident pressure (“safe distance”) is calculated by taking the product of the factor 45 and the cube root of the equivalent mass of TNT (Regulatory Guide 1.91).

The evaluation is based on a vapor-filled 6,644 gallon hydrazine tank. For the assumed atmospheric conditions, a heat of combustion of 8,345 Btu/lb, and a vapor specific gravity of 1.1, the mass of flammable hydrazine in the tank is 79 pounds. The resulting equivalent mass of TNT is calculated to be 330 pounds, and the resulting "safe distance" is 311 feet.

Results from the TNT equivalency method indicate that there are no potential explosive vapor threats from hydrazine storage tanks to safety related structures or operators in the Units 3 and 4 control rooms.

Insert the following subsection heading before the third paragraph of ESPA SSAR Subsection 2.2.3.2.3.

VEGP COL 2.2-1
VEGP COL 6.4-1
VEGP ESP COL 2.2-2

2.2.3.2.3.2 Other Chemical Hazards from Onsite Storage Tanks

VEGP ESP VAR 2.2-1

Replace the paragraph of new Subsection 2.2.3.2.3.2 with the following new paragraph.

Table 6.4-201 provides specific information about the chemicals described in DCD Table 6.4-1. This includes chemical names or limiting types and quantities. Except as noted, these chemicals have been suggested by Westinghouse for use in the AP1000 and have been evaluated in conjunction with AP1000 standard design and found not to present a hazard to the control room operators or to safety-related systems, structures, or components. In some instances, alternative chemicals to those proposed by Westinghouse have been suggested. These chemicals are comparable in function to those proposed by Westinghouse and are the same as those already in use for similar applications in VEGP Units 1 and 2. These chemicals also have been evaluated and found not to present a hazard to the control room operators or to safety-related systems, structures, or components. Therefore, no further analysis is required.

2.2.3.3 Fires

Add the following after the last paragraph in ESPA SSAR Subsection 2.2.3.3.

VEGP COL 2.2-1
VEGP COL 6.4-1

The specific application to Units 3 and 4 of these forest and industrial fire evaluations is further described below.

2.2.3.3.1 Forest Fires

The surrounding plant terrain is characterized by gently rolling hills and is approximately 30-percent farmland, with the remainder primarily wooded areas. The nearest forest to the Units 1 and 2 control room is the Sandhill-Upland hardwood pine forest with an assumed total area of approximately 3,169 acres and an assumed distance of 1,836 feet away. Based on historical data on forest fires from the state of Georgia, the average size of a forest fire typically is approximately 11.4 acres. The rate of spread is conservatively assumed to be 8 feet per minute with a duration of 4 hours.

The toxic chemicals emitted from a forest fire are CO, NO_2, and CH_4. The emission concentrations in the control room air intake were calculated using the infinite line source diffusion equation with the wind direction perpendicular to the line source and blowing directly toward the control room intake, and the Briggs plume rise equation, which accounts for the buoyancy effect from the heat of the fire. For Units 1 and 2, calculations were performed to demonstrate that the pollutant concentrations outside the control room air intake for a variety of wind speeds (from 0.25 to 10 m/sec) and the Pasquill stability category G are effectively zero. Therefore, the release of toxic combustion products from the onsite forest fire did not pose a hazard to the Units 1 and 2 control room operators.

Using the methodology described in NUREG/CR-1748, the heat flux and resultant temperature rise on plant structures due to a forest fire were also evaluated for Units 1 and 2. The calculated temperature rise (~46.5°C) is less than the allowable temperature rise (bulk 194°C and local 361°C). Therefore, a forest fire will not cause thermal damage to VEGP safety-related structures, based on the distance from the forest.

The centerline of VEGP Units 3 and 4 is approximately 2,100 feet west and 400 feet south of the center of the Unit 2 containment building. The Unit 4 containment is approximately 800 feet west of the Unit 3 containment. It is assumed that the distance from the nearest forest to VEGP Units 3 and 4 is the same as that from the forest to VEGP Units 1 and 2. Since Units 3 and 4 are approximately adjacent to Units 1 and 2 and the vegetation in the vicinity remains the same even after revegetation of the Units 3 and 4 construction site, the toxic chemicals emitted from a forest fire and the emission concentrations in the control room would have the same effect for Units 3 and 4. Therefore, the release of toxic combustion products from the onsite forest fire does not pose a hazard to the Units 3 and 4 control room operators.

2.2.3.3.2 Fire Due to an Accident at Offsite Industrial Storage Facility

Georgia Power Company's combustion turbine plant (Plant Wilson) is located approximately 1,350 meters from the VEGP Units 1 and 2 control room. Of the chemicals and toxic substances stored at this location, diesel fuel oil and miscellaneous oils are flammable. Based on a previous evaluation, a diesel fuel oil fire at Plant Wilson bounds the impacts from any fires of miscellaneous oils stored at Plant Wilson. One of the three tanks containing no. 2 diesel fuel oil is assumed to burn. The entire tank volume of 3×10^6 gallons is spilled into a dike area of 8,756 m^2.

The primary products of combustion emitted from a diesel fuel oil fire at Plant Wilson are CO, CO_2, CH_4, NO_2, SO_2, and SO_3. The toxicity limits in ppm for these constituents are 50 (CO), 5,000 (CO_2), 1.43×10^5 (CH_4), 2 (SO_2 and SO_3), and 3 (NO_2). Using the Briggs plume rise equations and by assuming the maximum burning rate of 0.12 inches/min, the maximum emission rate, duration of fire (8 hours), class A stability, and wind speeds (0.25-10 m/s), it was determined that the resulting concentrations of the primary products of combustion outside the Units 1 and 2 control room air intakes would not approach the above listed toxicity limits.

Using the methodology described in NUREG/CR-1748, the heat flux and resultant temperature rise on the VEGP structures due to a diesel fuel oil fire at Plant Wilson were also evaluated for the Units 1 and 2 control rooms. The calculated temperature rise (115°C) is less than the maximum allowable temperature rise (bulk 194°C and local 361°C). Since a fire at Plant Wilson is limiting (the largest source at the closest distance to the VEGP site), it is concluded that source fires and vapor cloud fires resulting from a delayed ignition at nearby industrial facilities will not cause thermal damage to safety-related structures at VEGP Units 1 and 2.

Units 3 and 4 are located at a farther distance from Plant Wilson than Units 1 and 2. Drawing from the conclusion based on the previous evaluation of Units 1 and 2, any industrial fire due to diesel oil or miscellaneous oils stored at Plant Wilson would not have an impact on control room habitability or cause thermal damage to safety-related structures at Units 3 and 4.

2.2.3.4 Radiological Hazards

Insert the following paragraph after the first paragraph in ESPA SSAR Subsection 2.2.3.4.

VEGP COL 2.2-1 The effect on the control rooms of VEGP Unit 3 and 4 of a postulated design basis accident (DBA) in Unit 1 or 2 was evaluated based on a LOCA in Unit 1 or 2, at uprated conditions, using the releases produced from the alternate source term (AST) methodology. The dose at the Unit 3 and 4 control rooms were determined considering the time-dependent source terms, the atmospheric dispersion factors (χ/Q values), the assumed occupancy rates, the volume of the control room, the HVAC filtration and flow rates, and the operator breathing rates. The χ/Q values from the containment of Unit 2 to the Units 3 and 4 control room air intakes were conservatively calculated using the same methodology and meteorology as was used to calculate the control room χ/Q values presented in Section 2.3.4. Breathing rates were assumed to be constant for the control room operators for the duration of the period evaluated. The occupancy rate in the control room was assumed to be 100 percent for the first 24 hours and then decreasing to 60 percent for the next 3 days and then to 40 percent over the remainder of the 30 day period. The resultant dose from this analysis is comparable to the dose reported in DCD Tier 2, Table 15.6.5-3 for a postulated LOCA in the AP1000 and is less than the GDC 19 limits.

VEGP DEP 1.1-1

2.2.4 COMBINED LICENSE INFORMATION FOR IDENTIFICATION OF SITE-SPECIFIC POTENTIAL HAZARDS

VEGP COL 2.2-1 This COL item is addressed in Subsections 2.2.3.2.3.1, 2.2.3.2.3.2, 2.2.3.3, 2.2.3.4, and ESPA SSAR Section 2.2.

Add the following new reference in the Section 2.2 References list.

VEGP SUP 2.2-1

201. Murphy, K.G., and K.M. Campe, “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19,” U.S. Atomic Energy Commission, 13th Air Cleaning Conference, 1974.

202. U.S. Environmental Protection Agency, “ALOHA (Areal Location of Hazardous Atmospheres),” Version 5.4.1, February 2007.

2.3 METEOROLOGY

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

This section of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

2.3.1.3.4 Precipitation Extremes

Insert the following after the last paragraph of ESPA SSAR Subsection 2.3.1.3.4.

VEGP SUP 2.3-1 The AP1000 safety-related roofs are sloped and designed to handle winter snowpack with margin to handle rainfall on top of the snowpack. The AP1000 design basis snow load of 75 psf (ground) and 63 psf (roof) has sufficient margin to include the weight of rain water adding to a pre-existing snow pack. Using ASCE 7-98 the design snow load 50 psf (ground) converts to 42 psf (roof). Therefore, the AP1000 design includes a 21 psf (63 - 42) margin above the design ASCE 7-98 requirement. This margin could accommodate the equivalent weight of 4" of water within the snow on the roof.

Winter PMP loads in excess of this loading are not considered credible based on the design of the roof. The safety related roofs are constructed of 15" thick reinforced concrete supported by steel beams. The roofs will not deflect enough to hold water under the snow load; therefore, ponding of rain water with pre-existing snow pack conditions will not occur. The physical arrangement of the AP1000 sloped roof is designed such that the 100-year snow pack will not prevent the winter PMP water from draining off the sloped roof system.

In addition the AP1000 roof includes R10 insulation that assures uniform temperatures on the roof surface. This minimizes the potential for ice dams that are typically formed across roofs with a temperature differential.

For the VEGP site, the 100 year snow load is 10 psf which is well within the 63 psf design basis snow load of the AP1000. Thus, for the VEGP site, a 53 psf margin is available to accommodate winter PMP water that may be impounded in the 100-year snow pack as the water flows off of the roof.

2.3.1.4 Meteorological Data for Evaluating the Ultimate Heat Sink

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Insert the following after the last paragraph of ESPA SSAR Subsection 2.3.1.4:

VEGP ESP COL 2.3-1 A reactor design has been chosen as specified in Section 1.1 that does not use an ultimate heat sink cooling tower to release heat to the atmosphere following a loss of coolant accident; therefore, evaluation of meteorological site characteristics such as maximum evaporation and drift loss and minimum water cooling conditions used to evaluate this design is not necessary.

2.3.1.5 Design Basis Dry- and Wet-Bulb Temperatures

The third from last and second from last paragraphs of ESPA SSAR Subsection 2.3.1.5 will be replaced with the following paragraph as shown:

VEGP ESP VAR 2.3-1 The AP1000 DCD maximum and minimum normal temperature site characteristics are 1-percent (99-percent) seasonal exceedance values. According to the ASHRAE 2001 Fundamentals Handbook, these are approximately equivalent to the annual 0.4-percent (99.6-percent) annual exceedance values. Thus, the maximum normal dry bulb temperature (1% seasonal exceedance) is 97° F with a coincident maximum normal wet bulb temperature of 76°F. The maximum normal non-coincident wet bulb temperature is 79°F. Additionally, the minimum normal dry bulb temperature (99% seasonal exceedance) is 21°F.

Insert the following new subsection after ESPA SSAR Subsection 2.3.3.3.

VEGP COL 2.3-3

2.3.3.4 VEGP Meteorological Monitoring Program Compliance

The meteorological monitoring program operated in support of VEGP Units 1 and 2 will also support the operation of VEGP Units 3 and 4. Characteristics of this monitoring program, include:

- siting of the meteorological tower with respect to potential obstructions to air flow (e.g., containment structures, cooling towers, tree lines),
- descriptions of the meteorological instrumentation (e.g., performance specifications, methods and equipment for recording sensor output, QA program for sensors and recorders, and data acquisition and reduction procedures), and
- operation, maintenance, and calibration procedures.

The NRC evaluated the meteorological monitoring program as part of the ESPA SSAR safety evaluation site audit on December 6, 2006 and through their review of ESPA SSAR Subsection 2.3.3.

The current monitoring program and its implementation were determined to meet the guidance in Proposed Revision 1 to Regulatory Guide 1.23 and found to provide an acceptable basis for estimating atmospheric dispersion conditions for accidental and routine releases of radioactive material to the atmosphere.

2.3.4 SHORT-TERM DIFFUSION ESTIMATES

Insert the following text at the beginning of ESPA SSAR Subsection 2.3.4.

VEGP COL 2.3-4 This subsection addresses the determination of conservative, short-term atmospheric dispersion estimates due to postulated design-basis, accidental releases of radioactive material to the ambient air for receptors located:

- on the Exclusion Area Boundary (EAB) and the outer boundary of the Low Population Zone (LPZ) (ESPA SSAR Subsections 2.3.4.1 and 2.3.4.2) to support the evaluation of offsite radiological consequences; and
- at air intake points to the control room (Subsection 2.3.4.3) to support the evaluation of personnel exposures inside the control room and the design of the control room habitability system.

This subsection also briefly addresses the determination of accident-related concentrations at the control room due to onsite and/or offsite airborne releases of hazardous materials such as flammable vapor clouds, toxic chemicals, and smoke from fires (Subsection 2.3.4.4).

In the AP1000 reactor DCD, the terms “site boundary” and “exclusion area boundary” are used interchangeably. Thus, the χ/Q value specified for the site boundary applies whenever a discussion in the DCD refers to the exclusion area boundary. In the ESPA SSAR Subsections 2.3.4.1 and 2.3.4.2 site specific χ/Q calculations, the term “Dose Calculation EAB” is equivalent to the DCD term “EAB”.

Short-term, dispersion-related site parameters at the site boundary and the LPZ boundary, on which the AP1000 design is based, are identified in DCD Tier 1, Table 5.0-1, DCD Tier 2, Table 2-1, and DCD Tier 2, Table 15A-5. As indicated above, site-specific dispersion characteristics that correspond to these site parameters are presented in ESPA SSAR Subsections 2.3.4.1 and 2.3.4.2.

Short-term, dispersion-related site parameters at the control room, also incorporated in the AP1000 design, are identified in DCD Tier 1, Table 5.0-1, DCD Tier 2, Table 2-1, and DCD Tier 2, Table 15A-6. Site-specific dispersion characteristics that correspond to these site parameters are presented in Subsection 2.3.4.3.

Tables 2.0-201 and 2.0-202 compare the applicable site parameters and corresponding site-specific characteristic values.

Insert the following text after the summary of PAVAN χ/Q Results at the end of ESPA SSAR Subsection 2.3.4.2.

VEGP COL 2.3-4 Using the same assumptions and methodology as described in the ESPA SSAR (which relied on DCD Revision 15), the short-term (accidental release) dispersion estimates at the EAB and the LPZ boundary were evaluated using the revised building dimensions provided in DCD Revision 17. That evaluation confirmed that the χ/Q values for the EAB and LPZ remain the same. This result is reasonable given that the designated receptor points at the EAB and the LPZ boundary are beyond the distance that would be influenced by building wake.

2.3.4.3 Radiological Accident Dispersion Estimates at the Control Room

Subsection 2.3.4.3.1 describes the dispersion modeling analysis used to determine short-term, relative concentration estimates associated with a postulated design-basis, accidental release of radioactive material to the atmosphere. The results of this dispersion analysis for receptors at air intake points to the control room are summarized in Subsection 2.3.4.3.2.

2.3.4.3.1 Regulatory Basis and Technical Approach

General Design Criterion 19 (*Control Room*) under 10 CFR Part 50, Appendix A, requires that the control room remain functional so that actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain the plant in a safe state under accident conditions.

Regulatory Guide 1.194, *Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants*, June 2003, provides guidance on utilizing the ARCON96 dispersion model to characterize atmospheric dispersion conditions (χ/Q values) that are input to the evaluation of the consequences of accidental airborne radiological releases on control room habitability. The ARCON96 dispersion model is described in NUREG/CR-6331 (*Atmospheric Relative Concentrations in Building Wakes*, PNNL-10521, Revision 1, May 1997). [Reference 201]

Five consecutive calendar years (from 1998 through 2002) of sequential hourly meteorological data, from the onsite monitoring program operated in support of

VEGP Units 1 & 2, were input to ARCON96 in model-required format. As such, the estimated χ/Q values represent the composite 5-year period of record. Wind data from both the 10- and 60-m measurement levels were included. Wind speed units of measure were in meters per second.

Joint data recovery of atmospheric stability class and 10-m level wind speed and wind direction was greater than 94 percent for each of the five years. Data recoveries for 60-m level wind data exceeded 95 percent for wind speed during each year, and ranged from about 93 to 97 percent for wind direction for all years except 1998 (at slightly more than 88 percent). Subsections 2.3.2 and 2.3.3 establish that these data are representative of site dispersion characteristics.

χ/Q values were estimated at two air intake points leading to the control room—at the Heating Ventilation and Air Conditioning (HVAC) system intake and at the annex building access door (i.e., the pathway for outside air to the control room is that due to building ingress/egress). These two air intake points, designated as Receptors 1 and 2, respectively, are illustrated in DCD Figure 15A-1.

These receptors may be contaminated by accidental radiological releases from any of eight potential sources (the two- or four-letter Source Indicator is included in the ARCON96 model):

- plant vent (Source Indicator - PV);
- passive containment cooling system (PCS) air diffuser (Source Indicator - AD);
- fuel building blowout panel (Source Indicator - BP);
- fuel building rail bay door (Source Indicator - BD);
- a steam vent (or line) break (Source Indicator - SV);
- Power Operated Relief Valves (PORV) and safety valves (Source Indicator - PORV);
- condenser air removal stack (Source Indicator - AR); and
- the containment shell (Source Indicator - CS).

These potential release points, designated as Sources 1 to 8, respectively, are also illustrated in DCD Figure 15A-1. Note that Source 4, the fuel building rail bay door in the list above, is referred to as the “Radwaste Building Truck Staging Area Door” in DCD Figure 15A-1.

The receptor locations are also reflected in the ARCON96 model and may be distinguished by the respective two-letter indicators “CR” (i.e., control room HVAC intake) and “AN” (annex building access door).

The release types used in the ARCON96 modeling analyses follow those specified in DCD Tier 2, Chapter 15, Appendix 15A. DCD Figure 15A-1 shows that among the potential release sources, the containment shell is considered to be a diffuse area source. All other releases are considered to be point sources.

The Regulatory Position in Section 3.2.2 of Regulatory Guide 1.194 specifies that the stack release mode in ARCON96 is appropriate for releases from a freestanding, vertical, uncapped stack that is outside the directionally dependent zone of influence of adjacent structures. Furthermore, Regulatory Guide 1.194 states that such a stack should be more than 2-1/2 times the height of adjacent structures. From DCD Table 15A-7, the height of the plant vent is 55.7 m above grade; the condenser air removal stack only 38.4 m above grade. Given that the PCS air diffuser sits atop the containment shield building at an elevation of 69.8 m above grade, the vertical criterion for stack releases is not met. Therefore, modeling these sources in stack release mode was not considered.

The Regulatory Position in Section 3.2.3 of Regulatory Guide 1.194 states that modeling sources using the vent release mode “may not be sufficiently conservative for accident evaluations” and so “should not be used in design basis assessments”. As neither a release from the condenser air removal stack nor the plant vent can be represented as stack releases, both potential sources were considered to be ground-level releases in the ARCON96 modeling analyses.

Different building cross-sectional areas were input to the model depending on the receptor being evaluated. For the annex building access door, a building cross-sectional area of 2,636 m^2 was used. This receptor, at an assumed elevation of 1.5 m, is located in a region where the air flow is under the influence of the combined structural wakes generated by the entire containment shield building, the auxiliary building, and the annex building. However, for this modeling analysis, the wake effects induced by the auxiliary building and the annex building were not considered. By excluding these two structures, the total building cross-sectional area is reduced, which is a relatively conservative assumption in that a smaller cross-sectional area results in higher χ/Q values.

The 2,636 m^2 cross-sectional area is based on an assumed diameter of the containment shield building of 43.3 m and an effective structural height of 60.9 m. The assumed diameter of the containment is slightly smaller than the actual diameter and is conservative since the smaller diameter results in a higher χ/Q. The effective structural height takes into account the fact that the containment shield building is a tapered structure beginning at elevation 170.84 ft above grade. The overall height of this building is 228.75 ft above grade. The effective structural height is taken, then, as the mid-point between the start of the taper and the overall building height—that is, 199.8 ft or 60.9 m.

For the receptor at the control room HVAC system intake, a cross-sectional area of 1,805 m^2 was assumed. This receptor, at an elevation of 19.9 m above grade, is located within the wake generated by that portion of the containment shield building that extends above the roof of the auxiliary building where this receptor is situated. The difference between the effective structural height of the containment shield building (i.e., 60.9 m, as discussed above) and the roof height of this part of the auxiliary building (i.e., 19.2 m above grade) is multiplied by the diameter of the containment shield building (i.e., 43.3 m) to yield the cross-sectional area input to the ARCON96 model for estimating χ/Q values at this receptor.

Specification of initial diffusion coefficients is only applicable to a hypothetical release from the containment shell which was modeled as a diffuse area source, as indicated previously. The Regulatory Positions in Sections 3.2.4.4 and 3.2.4.5 of Regulatory Guide 1.194 indicate that in the absence of site-specific empirical data, as is the case here, the initial horizontal and vertical diffusion coefficients may be estimated as follows:

- Sigma-y_o = Area Source Width ÷ 6; and
- Sigma-z_o = Area Source Height ÷ 6.

Consistent with those regulatory positions, the area source width and height are based on the horizontal and vertical dimensions used to determine the building cross-sectional areas input to the ARCON96 modeling analyses. For the receptor at the annex building access door, Sigma-y_o and Sigma-z_o are estimated to be 7.2 m (i.e., 43.3 m ÷ 6) and 10.2 m (i.e., 60.9 m ÷ 6), respectively. For the receptor at the control room HVAC intake, Sigma-y_o and Sigma-z_o are estimated to be 7.2 m (i.e., 43.3 m ÷ 6) and 7.0 m (i.e., 41.7 m ÷ 6), respectively.

Other parameters input to ARCON96 that are based on the recommendations in Regulatory Guide 1.194, Table A-2 (which are different, in some cases, than the default values in the model user's guidance, Reference 201) include:

- Surface Roughness Length = 0.2 (rather than the model default value of 0.1);
- Averaging Sector Width Constant = 4.3 (rather than the model default value of 4.0);
- Vertical Velocity, Stack Radius, and Stack Flow = 0 (all sources are assumed to be ground-level releases and so vertical velocity and stack radius are not used; stack flow during the course of an accident cannot be demonstrated with reasonable assurance);
- Release Height Elevation Difference = 0 (differences in grade elevations between all sources and receptors are only a few feet or less); and
- Wind Direction Window = 90 (default value in both Regulatory Guide 1.194 and Reference 201).

Finally, DCD Table 15A-7 lists the heights of the two modeled receptors and the eight potential sources of radioactive releases, the straight-line distances between these sources and the respective receptors.

2.3.4.3.2 ARCON96 Modeling Results

The χ/Qs determined by the ARCON96 dispersion model represent 95th-percentile values based on all of the hourly relative concentrations calculated using the 5-year meteorological data set input to the model. χ/Q values at the control room HVAC intake and at the annex building access door for time averaging intervals of 0-2 hours, 2-8 hours, 8-24 hours, 1-4 days, and 4-30 days are summarized in Tables 2.3-201 and 2.3-202, respectively.

2.3.4.4 Dispersion Estimates Associated with Accidental Onsite and Offsite Hazardous Material Releases

Potential control room habitability effects and personnel exposures at VEGP Units 3 & 4 due to:

- postulated accidental releases of chemicals and other hazardous materials stored onsite, and at offsite locations within 5 miles of the units;
- for toxic or flammable materials carried over nearby transportation routes (e.g., roadways, railways, and waterways); and
- explosions

were addressed in Subsection 2.2.3 and in ESPA SSAR Section 2.2.

Concentrations at the control room HVAC intake and at the annex building access door due to accidental hazardous chemical releases were determined and evaluated in consideration of the guidance in Regulatory Guide 1.78, *Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release*, Revision 1, December 2001.

2.3.5 LONG-TERM DIFFUSION ESTIMATES

Insert the following text after the last paragraph of ESPA SSAR Subsection 2.3.5.

VEGP COL 2.3-5 In the AP1000 reactor DCD, the terms “site boundary” and “exclusion area boundary” (EAB) are used interchangeably. Thus, the χ/Q specified for the site boundary applies whenever a discussion in the DCD refers to the exclusion area boundary. In ESPA SSAR Subsection 2.3.5 site specific χ/Q calculations, the term “Dose Calculation EAB” is equivalent to the DCD term “EAB”.

Using the same assumptions and methodology as described in the ESPA SSAR (which relied on DCD Revision 15), along with the building dimensions provided in DCD Revision 17, the long-term (routine release) dispersion and deposition estimates were evaluated at the Dose Calculation EAB and at the various receptor locations described in the ESPA SSAR. This evaluation confirmed that the χ/Q values for the EAB and the various receptor locations are within approximately 3.3% of those provided in the ESPA. This result is reasonable given that the designated receptor points at the EAB and the various receptor locations are beyond the distance that would be appreciably influenced by building wake.

2.3.6 COMBINED LICENSE INFORMATION

2.3.6.1 Regional Climatology

VEGP COL 2.3-1 This COL item is addressed in ESPA SSAR Subsection 2.3.1

2.3.6.2 Local Meteorology

VEGP COL 2.3-2 This COL item is addressed in ESPA SSAR Subsection 2.3.2

2.3.6.3 Onsite Meteorological Measurements Program

VEGP COL 2.3-3 This COL item is addressed in Subsection 2.3.3.4 and ESPA SSAR Subsection 2.3.3

2.3.6.4 Short-Term Diffusion Estimates

VEGP COL 2.3-4 This COL item is addressed in Subsections 2.3.4, 15.6.5.3.7.3, Appendix 15A.3.3, and ESPA SSAR Subsection 2.3.4

2.3.6.5 Long Term Diffusion Estimates

VEGP COL 2.3-5 This COL item is addressed in Subsection 2.3.5 and ESPA SSAR Subsection 2.3.5

Add the following information after DCD Subsection 2.3.6.5

VEGP SUP 2.3-2 2.3.7 REFERENCES

201. NUREG/CR-6331, *Atmospheric Relative Concentrations in Building Wakes*, PNNL-10521, Revision 1, May 1997.

VEGP COL 2.3-4

Table 2.3-201
ARCON96 X/Q Values at the Control Room HVAC Intake

Release Point	0 – 2 hours	2 – 8 hours	8 – 24 hours	1 – 4 days	4 – 30 days
Plant Vent	**2.02E-03**	**1.58E-03**	**6.37E-04**	**5.12E-04**	**3.82E-04**
PCS Air Diffuser	**1.68E-03**	**1.29E-03**	**5.47E-04**	**4.55E-04**	**3.34E-04**
Fuel Building Blowout Panel	**1.54E-03**	**1.11E-03**	**4.42E-04**	**3.57E-04**	**2.59E-04**
Fuel Building Rail Bay Door	**1.15E-03**	**8.29E-04**	**3.35E-04**	**2.62E-04**	**1.86E-04**
Steam Line Break	**1.48E-02**	**1.20E-02**	**5.41E-03**	**3.93E-03**	**3.26E-03**
PORV & Safety Valves	**1.31E-02**	**1.02E-02**	**4.62E-03**	**3.29E-03**	**2.77E-03**
Condenser Air Removal Stack	**1.54E-03**	**1.17E-03**	**5.36E-04**	**3.94E-04**	**2.78E-04**
Containment Shell (As Diffuse Area Source)	**3.20E-03**	**1.82E-03**	**8.27E-04**	**7.22E-04**	**5.70E-04**

VEGP COL 2.3-4

Table 2.3-202
ARCON96 X/Q Values at the Annex Building Access Door

Release Point	**0 – 2 hours**	**2 – 8 hours**	**8 – 24 hours**	**1 – 4 days**	**4 – 30 days**
Plant Vent	**4.32E-04**	**3.52E-04**	**1.44E-04**	**1.15E-04**	**8.47E-05**
PCS Air Diffuser	**4.48E-04**	**3.38E-04**	**1.44E-04**	**1.17E-04**	**8.77E-05**
Fuel Building Blowout Panel	**3.77E-04**	**2.84E-04**	**1.18E-04**	**9.50E-05**	**6.83E-05**
Fuel Building Rail Bay Door	**3.48E-04**	**2.60E-04**	**1.09E-04**	**8.75E-05**	**6.16E-05**
Steam Line Break	**9.23E-04**	**7.31E-04**	**2.98E-04**	**2.37E-04**	**1.75E-04**
PORV & Safety Valves	**9.81E-04**	**7.69E-04**	**3.12E-04**	**2.49E-04**	**1.87E-04**
Condenser Air Removal Stack	**4.00E-03**	**3.15E-03**	**1.35E-03**	**1.04E-03**	**8.05E-04**
Containment Shell (As Diffuse Area Source)	**3.93E-04**	**3.16E-04**	**1.32E-04**	**1.07E-04**	**8.14E-05**

2.4 HYDROLOGIC ENGINEERING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

This section of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

VEGP DEP 1.1-1 Subsection 2.4.1 of the DCD is renumbered as Subsection 2.4.15. This is being done to accommodate the incorporation of Regulatory Guide 1.206 numbering conventions for Section 2.4.

2.4.2 FLOODS

2.4.2.3 Effects of Local Intense Precipitation

Add the following to the end of ESPA SSAR Subsection 2.4.2.3.

VEGP COL 2.4-2 Based on work subsequent to the submittal of the referenced ESPA SSAR, the design elements of the VEGP Units 3 and 4 storm water management system pertaining to the local PMP flood event are described below.

As shown in Figure 2.4-201, the VEGP Units 3 and 4 power block is graded to direct runoff east and west to three north-south ditches which will outfall to the concrete-lined main ditch, running east and west for 2,000 feet along the south side of the power block. The trapezoidal ditch cross section has a 10-foot bottom width with 2:1 side slopes, sized to provide adequate conveyance for PMP discharges. At the southwest corner of the power block, the main ditch turns due south, and the bottom width is increased to 14 feet. From the west, it intercepts runoff from the construction laydown area; from the east it intercepts discharge from three ditches draining the cooling tower block.

The main ditch has a mild slope (0.22%) for its first 3,800 feet, at which point the slope increases to over 5% before outfalling about 4,500 feet from its upstream end into Debris Basin No. 2, which drains to an unnamed tributary of Daniels Branch, about 2,500 feet upstream of Telfair Pond.

The main ditch drains runoff from a total area of about 473 acres during the PMP event, including about 80 acres from the VEGP Units 3 and 4 power block, 97 acres from the cooling tower area, 56 acres south of the cooling tower, and 82 acres from the laydown area. An additional 133 acres from an area north of the haul road and 25 acres from VEGP Units 1 and 2 power block are assumed to drain to the main ditch for the PMP design rainfall event due to blocked culverts.

The local PMP event was modeled in HEC-HMS, which is an industry standard program for this application. For inputs of rainfall and drainage basin characteristics, the program outputs stream flow hydrographs at selected locations within the drainage basin (Reference 203).

The design rainfall hyetograph was developed in HEC-HMS utilizing the frequency storm option in the Meteorologic Models module (Reference 204). This option requires the input of PMP point depths for durations of 5, 15, 60, 120, 180, and 360 minutes.

Based on the logarithmic fit to the data shown in ESPA SSAR Table 2.4.2-3, a PMP total depth was estimated for the missing durations, as indicated in Table 2.4-201. An intensity position of 50 percent was selected for the HEC-HMS calculation, consistent with the alternating block pattern used in standard analysis (Reference 201). The rainfall hyetograph developed from the data is shown in Figure 2.4-202.

Elements within the HEC-HMS basin model include subbasins, reaches, and junctions. Runoff hydrographs were developed for subbasins and were routed through the channel system along reaches connected by junctions (Reference 204).

This calculation utilized the SCS Hydrograph Methodology (Reference 204), which requires the following parameters for each subbasin:

- Drainage Area, in square miles
- Runoff Curve Number and Initial Abstractions
- Lag Time, in minutes
- Base flow, in cfs

Drainage areas were delineated and measured for each subbasin shown in Figure 2.4-201.

The runoff curve number (CN) was selected as 98 for all types of cover to provide a conservative estimate of runoff volume and peak discharges and to account for nonlinear basin response to extreme rainfall events. Under normal flood conditions, the area-weighted average for each subbasin could be expected to vary between 50 and 75, while a CN value of 98 is typically used for impervious areas.

The lag time was estimated as 60 percent of the time of concentration, which is the time required for all areas of the drainage basin to be contributing to outflow. It was calculated for each subbasin as the sum of the overland, shallow concentrated, and channel flow times along the critical flow path through the basin using standardized equations (Reference 204).

An assumption of the PMP design storm is that a 50-percent PMP storm has occurred 3 days prior to the start of the rainfall associated with the actual PMP event, so some flow in the drainage ditches would be expected as the result of interflow draining from the pervious areas of the upstream watershed, although it would not be a significant quantity for this site, considering the limited drainage area. For this site, base flow is taken as zero for subbasins that are completely paved. Base flow is estimated on a 100 cfs per square mile basis for subbasins with uncovered ground.

The SCS unit hydrograph parameters calculated for each of the subbasins in the HEC-HMS models are provided in Table 2.4-202.

The subbasin hydrographs are added at junctions and routed through channel reaches. Straight lag time was used for the smaller reaches; the kinematic wave routing option for most of the main channel reaches. The routing parameters are shown in Table 2.4-203.

Peak discharges from all subbasins, at all junctions, and at the downstream end of each of the routing reaches resulting from modeling the PMP rainfall event in HEC-HMS are summarized in Table 2.4-204. The highlighted entries indicate junctions along the main ditch. The hydrographs simulated for these junctions are shown in Figure 2.4-203.

The backwater analysis for the PMP drainage network was developed in HEC-RAS (Reference 205). Cross sections were developed for the main drainage ditch and feeder channels with topographic data for the overbank area, using the proposed geometric configuration for the channels. The locations of the cross sections used in the HEC-RAS model are shown in Figure 2.4-201a.

The assumptions made and the data utilized in the development of the hydraulic model are as follows:

- All channels are concrete lined, so no local scoured-out cross sections are utilized in the model.
- All culverts in the model are assumed to be 100% blocked by debris collected from the catchment.
- The blocked culverts within the power block area are modeled as in-line weirs in HEC-RAS following common hydraulic engineering practice (Reference 205). The effect of the blocked culvert in Feeder Ditch 4 is

accounted for by adjusting cross section geometry to indicate the ditch is filled in at the culvert location.

- Peak discharges from the HEC-HMS model were used at all sections in a steady-state calculation. Based on the close coincidence in time of peak discharges along the main channel and in the contributing subbasins, as shown in Table 2.4-204 and Figure 2.4-202, this was considered to be a reasonable simplification.

Peak PMP discharges simulated in HEC-HMS at eight locations along the main channel (nodes M1 through M8, as shown in Figure 2.4-201) were utilized in HEC-RAS at the cross sections indicated in Table 2.4-205.

In HEC-HMS, discharge was calculated at two points along each of the feeder ditches 1, 2, and 3, within the power block area. To better represent the lateral inflow to the feeder ditches along their entire length, the discharge from the two HEC-HMS nodes for each ditch were distributed linearly to each section in the models of the respective ditches to better represent lateral inflow, as summarized in Table 2.4-206.

The model was run with the mixed flow regime option with the downstream boundary condition taken as normal depth at Section 45+00, with the energy slope equal to the channel slope at that point of 5 percent (section stationing is shown at 500-foot intervals in Figure 2.4-201). The upstream boundary conditions were also taken as normal depth with an energy slope of 0.0001 to account for the severe backwater effect at the upstream ends of the branches of the drainage system.

The Manning's n roughness values used in the model were selected for standard conservative assumptions (Reference 202) as follows:

- concrete feeder ditches (assumed to be well maintained) with n = .014 and overbank areas assumed to be gravel bottom and concrete curbs with n = .020

- all other ditches assumed to be float-finished concrete lining with n = .015 and overbank areas assumed to be short grass with n = .030

The results of the mixed-flow regime back water calculation for PMP discharges in the drainage network are presented in Table 2.4-207. Flow is supercritical in the steep reach of the main ditch from the downstream section up to section 37+00, with control (Froude No. = 1) at section 38+00, with a velocity of 16.6 fps and a depth of 14.14 feet. Velocities decrease and depths generally increase in the mild-sloped (S = .0022) reach upstream of that section to 3.7 fps and 15.98 feet respectively at section 20+00, and 0.9 fps and 11.98 feet respectively at section 1+00.

The feeder ditches draining the power block area are subject to high tailwater conditions in the main ditch for the PMP runoff event. The HEC-RAS output indicates that the maximum floodwater surface elevation would be between 219.28 ft msl in the SW corner and 219.47 ft msl in the NE corner of the VEGP Units 3 and 4 power block. As all safety-related facilities have entry elevations at or above 220 ft msl, it has been determined that the maximum local PMP flood elevation is at least 0.53 ft below any entry to any safety related facility, and the flooding of safety-related facilities due to this PMP event does not occur. Configuration control of the plant layout, as assumed in the hydraulic model described above, is governed by applicable plant procedures.

In summary, the main ditch system has been designed to convey the peak discharge of the PMP flood event safely offsite. In addition, site grading is sufficiently sloped to convey runoff overland from the local PMP event away from all buildings and safety-related equipment, without flooding.

The required maintenance for the drainage ditches and overbank areas will be determined during the quarterly walk-through inspections of the drainage features (main drainage and feeder ditches and their overbank areas) in the Units 3 and 4 portion of the protected area and from the protected area fence through the Units 3 and 4 cooling tower area.

2.4.10 FLOODING PROTECTION REQUIREMENTS

Add the following paragraphs at the end of ESPA SSAR Subsection 2.4.10.

VEGP COL 2.4-2 The maximum flood elevation in the Savannah River at the VEGP site is El. 178.10 ft msl, resulting from the cascading failure of upstream dams including wind setup and wave run-up, as discussed in ESPA SSAR Subsection 2.4.4. This elevation is well below the VEGP site grade at El. 220 ft msl.

Subsection 2.4.2 subsequently considered the flooding effects of local intense precipitation (also termed as the local probable maximum precipitation or local PMP) on the Units 3 and 4 safety-related structures at the VEGP site. A local PMP drainage analysis was performed by conservatively assuming that all underground storm drains and culverts were clogged. Details of the local PMP analysis and the resulting flood levels are presented in Subsection 2.4.2. As indicated in Subsection 2.4.2, the maximum water level in the Units 3 and 4 power block area due to the local PMP flood event is calculated to be at El. 219.47 ft msl. The entrances and openings for all safety-related facilities are located at or above the VEGP site grade of EL. 220 ft msl.

Thus, none of the VEGP Units 3 and 4 safety-related structures will be adversely affected by any flood event. Consequently, no flood protection measures are required for VEGP Units 3 and 4. Additionally, no technical specifications or emergency procedures to implement flood protection activities are required.

Furthermore, the design of VEGP Units 3 and 4 drainage facilities incorporates measures to ensure that VEGP Units 1 and 2 safety-related facilities are not subject to flooding during construction and operation of VEGP Units 3 and 4. Drainage from the VEGP Units 3 and 4 portion of the site during construction and operation of Units 3 and 4 is directed away from the drainage facilities of Units 1 and 2. Hence, drainage from the VEGP Units 3 and 4 site area do not affect the safety-related structures, systems, and components of VEGP Units 1 and 2.

As discussed in Subsection 3.4.1.1 of the AP1000 DCD (Tier 2), the roofs of all safety-related structures are designed to prevent flooding of, or leakage into, safety-related structures, systems, and components as a result of the PMP on the roofs. The design basis combination of a 100-year return period ground-level snowpack and 48-hour probable maximum winter precipitation, as applied to safety-related roofs, is discussed in ESPA SSAR Subsection 2.3.1.3.4.

2.4.12 GROUNDWATER

2.4.12.3 Monitoring of Safeguards Requirements

Add the following after the second paragraph in ESPA SSAR Subsection 2.4.12.3.

The existing SNC groundwater monitoring programs are evaluated with respect to placement of the new units in Subsection 2.4.12.3.1 below.

VEGP SUP 2.4-1 2.4.12.3.1 Long Term Groundwater Level Monitoring

ESPA SSAR Section 2.4.12.3 indicates that the existing groundwater monitoring programs would be evaluated to determine if any additional monitoring of existing observation wells or construction of new observation wells would be required to adequately monitor the impact on groundwater. The results of the evaluation indicate that the long term collection of Units 3 and 4 water table level data is appropriate to confirm the direction of groundwater flow in the vicinity of the power blocks of Units 3 and 4.

Groundwater level data will be collected from a network of observation wells similar to that utilized for the ESP phase (June 2005 through July 2007) to ensure the data is comparable to that of the ESP phase. Most of the active Units 1 and 2 observation wells that were included in the ESPA data will not be impacted by the earth moving activities for Units 3 and 4. However, most of the remainder of the ESPA observation wells will be impacted by these earthwork activities. The number and location of the replacement wells to be installed for long term monitoring will be determined after the earthwork activities are complete and heavy construction is well underway. Some of these observation wells will be installed in the Units 3 and 4 power block areas.

Groundwater level monitoring will be initiated prior to commercial operation of Unit 3 and revised as needed based upon the review and evaluation of the observed data.

Add the following new subsections after ESPA Subsection 2.4.13.

2.4.14 TECHNICAL SPECIFICATIONS AND EMERGENCY OPERATION REQUIREMENTS

VEGP COL 2.4-6 The plant elevation (220 ft MSL) of the VEGP is above the design basis river flood elevation and the probable maximum precipitation flood elevation; therefore, due to design there are no requirements for emergency protective measures designed to minimize the impact of hydrology-related events on safety-related facilities, and none are incorporated into the technical specifications or emergency procedures.

VEGP DEP 1.1-1

2.4.15 COMBINED LICENSE INFORMATION

2.4.15.1 Hydrological Description

VEGP COL 2.4-1 This COL item is addressed in ESPA SSAR Subsection 2.4.1.

2.4.15.2 Floods

VEGP COL 2.4-2 This COL item is addressed in Subsection 2.4.2, Subsection 2.4.10 and ESPA SSAR Subsections 2.4.2, 2.4.3, 2.4.4, 2.4.5, 2.4.6, and 2.4.10.

2.4.15.3 Cooling Water Supply

VEGP COL 2.4-3 This COL item is addressed in ESPA SSAR Subsection 2.4.12.

2.4.15.4 Groundwater

VEGP COL 2.4-4 This COL item is addressed in ESPA SSAR Subsection 2.4.12.

2.4.15.5 Accidental Release of Liquid Effluents into Ground and Surface Water

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VEGP COL 2.4-5 This COL item is addressed in ESPA SSAR Subsection 2.4.13.

2.4.15.6 Emergency Operation Requirement

VEGP COL 2.4-6 This COL item is addressed in Subsection 2.4.14.

2.4.16 REFERENCES

201. Chow, Maidment, and Mays, “Applied Hydrology,” McGraw-Hill, 1988.

202. Chow, Ven T., “Open Channel Hydraulics,” McGraw-Hill, 1959.

203. United States Army Corps of Engineers, “HEC-HMS Hydrologic Modeling System,” Version 3.0.1 User’s Manual, April 2006.

204. United States Army Corps of Engineers, “HEC-RAS River Analysis,” Version 3.0.1 Technical Reference Manual, March 2000.

205. United States Army Corps of Engineers, “HEC-RAS River Analysis System,” Version 3.1 User’s Manual, November 2002.

VEGP COL 2.4-2

Table 2.4-201
Rainfall Depths Used as Input for Frequency Storm HEC-HMS Module

Duration, Minutes	Depth, Inches
5	6.20[a]
15	9.80[a]
60	19.20[a]
120	23.52[b]
180	25.95[b]
360	31.00[a]

a) Calculated with HMR51/52

b) Calculated by curve fitting

VEGP COL 2.4-2

Table 2.4-202
Subbasin Parameters for Entry in Unit 3 & 4 Drainage System
HEC-HMS Model

Subbasin	Subbasin Area			Base flow?	Baseflow, cfs	Tc, min	Lag, min
	sq. ft	acres	sq. mi.	0:no, 1: yes			
FD1W	222,801	5.11	0.0080	0	0.0	13.9	8.4
FD2E	373,473	8.57	0.0134	0	0.0	26.4	15.8
FD2W	265,336	6.09	0.0095	0	0.0	24.8	14.9
FD3E	387,040	8.89	0.0139	0	0.0	10.8	6.5
FD3W	576,756	13.24	0.0207	1	2.1	22.2	13.3
FD5aN	555,561	12.75	0.0199	0	0.0	9.9	5.9
FD6aE	407,411	9.35	0.0146	1	1.5	18.7	11.2
FD6bE	872,411	20.03	0.0313	1	3.1	22.8	13.7
LD2	885,606	20.33	0.0318	1	3.2	13.7	8.2
LD3	295,463	6.78	0.0106	1	1.1	12.9	7.7
LD4	302,610	6.95	0.0109	1	1.1	12.0	7.2
LD5	133,962	3.08	0.0048	1	0.5	9.8	5.9
M1S	245,755	5.64	0.0088	0	0.0	9.1	5.5
M1W	210,975	4.84	0.0076	0	0.0	8.7	5.2
M2E	354,572	8.14	0.0127	0	0.0	10.6	6.3
M2S	650,674	14.94	0.0233	0	0.0	21.4	12.8
M2W	252,841	5.80	0.0091	0	0.0	8.9	5.4
M3E	406,105	9.32	0.0146	0	0.0	8.1	4.9
M3S	821,527	18.86	0.0295	0	0.0	15.6	9.3
M3W	578,010	13.27	0.0207	1	2.1	18.7	11.2
M4W	94,250	2.16	0.0034	1	0.3	15.3	9.2
M5W	289,232	6.64	0.0104	1	1.0	15.9	9.5
M6E	744,743	17.10	0.0267	0	0.0	11.3	6.8
M6W	394,537	9.06	0.0142	1	1.4	14.5	8.7
M7E	544,496	12.50	0.0195	1	2.0	20.1	12.1
M7W	290,945	6.68	0.0104	1	1.0	14.2	8.5
M8Cat	1,831,895	42.05	0.0657	1	6.6	37.6	22.6
M8W	326,470	7.49	0.0117	1	1.2	14.3	8.6
OF1	4,204,636	96.53	0.1508	1	15.1	45.5	27.3
OF2	1,597,617	36.68	0.0573	1	5.7	32.0	19.2
UN12-N	762,505	17.50	0.0274	0	0.0	25.1	15.1
UN12-S	758,482	17.41	0.0272	0	0.0	25.1	15.1
Totals:	20,638,697	473.80	0.7403	0	48.9		

VEGP COL 2.4-2

Table 2.4-203
Routing Parameters for Reaches in South-Side Drainage System Model

Direct Lag Reaches:

Reach	Length, ft.	vel., fps	lag. min.
FD1-M1	600	2	5.0
FD2-M2	600	1	10.0
FD3-M3	700	3.5	3.3
FD5a-M6	1150	3.5	5.5
FD6a-M7	1150	3.5	5.5
FD6b-a	900	3.5	4.3
M3-M4	200	5	0.7
OF1-FD3	N/A	*	23.9
OF2-FD3	N/A	**	10.0
RROF2-M3	775	2.1	6.2
RROF3-M5	600	2.1	4.8
RROF4-M6	600	2.1	4.8
RROF5-M7	600	2.1	4.8

* Use the time of concentration for subbasin FD3W for lag time.
** Use shallow concentrated flow velocity from FD3W for 350 feet and total ditch flow time.

Kinematic Wave Reaches:

Reach	Length, ft.	slope ft./ft.	n	# Subreaches	Shape	W, ft.	XH:1V
M1-M2	900	0.0022	0.015	2	TRAP	10	2
M2-M3	1,100	0.0022	0.015	2	TRAP	10	2
M4-M5	500	0.0022	0.015	2	TRAP	14	2
M5-M6	600	0.0022	0.015	2	TRAP	14	2
M6-M7	450	0.0022	0.015	2	TRAP	14	2
M7-M8	670	0.0526	0.015	2	TRAP	14	2

VEGP COL 2.4-2

Table 2.4-204 (Sheet 1 of 2)
Summary Results of HEC-HMS Model of Unit 3 & 4 PMP Drainage System

Element	Area, mi^2	Qpeak, cfs	Time of peak *	Runoff	
				inches	ac-ft
FD1	0.0354	698	01Jan2007, 03:15	30.68	57.9
FD1-M1	0.0354	698	01Jan2007, 03:20	30.68	57.9
FD1W	0.0080	200	01Jan2007, 03:10	30.68	13.1
FD2	0.0229	434	01Jan2007, 03:20	30.68	37.5
FD2E	0.0134	253	01Jan2007, 03:20	30.68	21.9
FD2-M2	0.0229	434	01Jan2007, 03:30	30.68	37.5
FD2W	0.0095	183	01Jan2007, 03:15	30.68	15.5
FD3	0.2427	2,924	01Jan2007, 03:45	34.19	442.5
FD3E	0.0139	380	01Jan2007, 03:10	30.68	22.7
FD3-M3	0.2427	2,924	01Jan2007, 03:50	34.19	442.5
FD3W	0.0207	426	01Jan2007, 03:15	34.45	38
FD5a	0.0199	547	01Jan2007, 03:10	30.68	32.6
FD5a-M6	0.0199	547	01Jan2007, 03:15	30.68	32.6
FD5aN	0.0199	547	01Jan2007, 03:10	30.68	32.6
FD6a	0.0459	902	01Jan2007, 03:20	34.41	84.2
FD6aE	0.0146	321	01Jan2007, 03:15	34.5	26.9
FD6a-M7	0.0459	902	01Jan2007, 03:25	34.41	84.2
FD6b	0.0313	635	01Jan2007, 03:15	34.36	57.4
FD6b-a	0.0313	629	01Jan2007, 03:20	34.36	57.4
FD6bE	0.0313	635	01Jan2007, 03:15	34.36	57.4
LD2	0.0318	808	01Jan2007, 03:10	34.42	58.4
LD3	0.0106	277	01Jan2007, 03:10	34.54	19.5
LD4	0.0109	291	01Jan2007, 03:10	34.43	20
LD5	0.0048	133	01Jan2007, 03:10	34.55	8.8
M1	0.0790	1,433	01Jan2007, 03:15	30.68	129.3
M1-M2	0.0790	1,420	01Jan2007, 03:20	30.7	129.3
M1S	0.0088	241	01Jan2007, 03:10	30.68	14.4
M1W	0.0076	207	01Jan2007, 03:10	30.68	12.4

* for an assumed start of storm time of 01 Jan 2007, 0:00 hours

VEGP COL 2.4-2

Table 2.4-204 (Sheet 2 of 2)
Summary Results of HEC-HMS Model of Unit 3 & 4 PMP Drainage System

Element	Area, mi^2	Qpeak, cfs	Time of peak *	Runoff	
				inches	ac-ft
M2	0.1470	2,546	01Jan2007, 03:15	30.69	240.6
M2E	0.0127	349	01Jan2007, 03:10	30.68	20.8
M2-M3	0.1470	2,530	01Jan2007, 03:15	30.7	240.7
M2S	0.0233	486	01Jan2007, 03:15	30.68	38.1
M2W	0.0091	249	01Jan2007, 03:10	30.68	14.9
M3	0.4863	6,291	01Jan2007, 03:15	32.84	851.8
M3E	0.0146	403	01Jan2007, 03:05	30.68	23.9
M3-M4	0.4863	6,291	01Jan2007, 03:15	32.84	851.8
M3S	0.0295	699	01Jan2007, 03:10	30.68	48.3
M3W	0.0207	455	01Jan2007, 03:15	34.45	38
M4	0.4897	6,367	01Jan2007, 03:15	32.85	858
M4-M5	0.4897	6,336	01Jan2007, 03:15	32.86	858.1
M4W	0.0034	81	01Jan2007, 03:10	33.96	6.2
M5	0.5107	6,835	01Jan2007, 03:15	32.92	896.7
M5-M6	0.5107	6,790	01Jan2007, 03:15	32.93	896.8
M5W	0.0104	244	01Jan2007, 03:10	34.25	19
M6	0.5824	8,463	01Jan2007, 03:15	32.81	1019.1
M6E	0.0267	723	01Jan2007, 03:10	30.68	43.7
M6-M7	0.5824	8,431	01Jan2007, 03:15	32.82	1019.3
M6W	0.0142	351	01Jan2007, 03:10	34.35	26
M7	0.6630	9,919	01Jan2007, 03:15	33.01	1167.3
M7E	0.0195	418	01Jan2007, 03:15	34.49	35.9
M7-M8	0.6630	9,896	01Jan2007, 03:15	33.01	1167.4
M7W	0.0104	260	01Jan2007, 03:10	34.25	19
M8	0.7404	11,021	01Jan2007, 03:15	33.16	1309.5
M8Cat	0.0657	1,058	01Jan2007, 03:25	34.41	120.6
M8W	0.0117	291	01Jan2007, 03:10	34.49	21.5
OF1	0.1508	2,203	01Jan2007, 03:30	34.4	276.7
OF1-FD3	0.1508	2,178	01Jan2007, 03:55	34.4	276.7
OF2	0.0573	993	01Jan2007, 03:20	34.38	105.1
OF2-FD3	0.0573	993	01Jan2007, 03:30	34.38	105.1
RROF2	0.0318	808	01Jan2007, 03:10	34.42	58.4
RROF2-M3	0.0318	763	01Jan2007, 03:15	34.42	58.4
RROF3	0.0106	277	01Jan2007, 03:10	34.54	19.5
RROF3-M5	0.0106	267	01Jan2007, 03:15	34.54	19.5
RROF4	0.0109	291	01Jan2007, 03:10	34.43	20
RROF4-M6	0.0109	278	01Jan2007, 03:15	34.43	20
RROF5	0.0048	133	01Jan2007, 03:10	34.55	8.8
RROF5-M7	0.0048	124	01Jan2007, 03:15	34.55	8.8
UN12-N	0.0274	522	01Jan2007, 03:15	30.68	44.8
UN12-S	0.0272	518	01Jan2007, 03:15	30.68	44.5

* for an assumed start of storm time of 01 Jan 2007, 0:00 hours

VEGP COL 2.4-2

Table 2.4-205
Location of Main Channel Discharge Points in HEC-RAS Model

Main Ditch Section	HEC-RAS Section	PMP Q, cfs	Comment	HEC-HMS node
0+00	46	759	Upstream end of model	M1S+UN12-S
0+60	45.1	1433	Confluence of Feeder Ditch 1	M1
10+00	36	2546	Confluence of Feeder Ditch 2	M2
20+00	26	6291	Confluence of Feeder Ditch 3	M3
22+00	24	6367	Confluence of Feeder Ditch 4	M4
27+00	19	6835	Local Inflow	M5
33+00	13	8463	Confluence of Feeder Ditch 5	M6
38+00	8	9919	Confluence of ditch (not modeled)	M7
45+00	1	11021	Local Inflow	M8

VEGP COL 2.4-2

Table 2.4-206
Location of Main Channel Discharge Points in HEC-RAS Model

Feeder Ditch	HEC-RAS Section	PMP Q, cfs	Comment	HEC-HMS node
1	83	103	Distribute discretized HEC-HMS node flow between all HEC-RAS sections	
1	82	206		
1	81	309		
1	80	413		
1	79	516		
1	78	619		
1	77	722	Q = FD1W + UN12-N	FD1
1	76	843	Distribute discretized HEC-HMS node flow between all HEC-RAS sections	
1	75	964		
1	74	1085		
1	73	1205		
1	72	1326		
1	71	1447	Q = M1W + UN12-S + FD1W + UN12-N	U/S of M1
2	83	62	Distribute discretized HEC-HMS node flow between all HEC-RAS sections	
2	82	124		
2	81	186		
2	80	249		
2	79	311		
2	78	373		
2	77	435	Q = FD2W + FD2E	FD2
2	76	535	Distribute discretized HEC-HMS node flow between all HEC-RAS sections	
2	75	634		
2	74	734		
2	73	834		
2	72	933		
2	71	1033	Q = M2W + M2E + FD2W + FD2E	U/S of M2
3	83	568	Distribute discretized HEC-HMS node flow between all HEC-RAS sections	
3	82	1136		
3	81	1705		
3	80	2273		
3	79	2841		
3	78	3409		
3	77	3978	Q = FD3W + FD3E + OF1-FD3 + OF2-FD3	FD3
3	76	4172	Distribute discretized HEC-HMS node flow between all HEC-RAS sections	
3	75	4366		
3	74	4561		
3	73	4755		
3	72	4949		
3	71	5144	Q = Sum(FD3)+ M3E + RROF2-M3	U/S of M3

VEGP COL 2.4-2

Table 2.4-207 (Sheet 1 of 3)
Summary of HEC-RAS output for PMP Profile

HEC-RAS identification		Section	Q Total	Min Ch El	Bottom	W.S. Elev	Crit W.S.	E.G. Elev	Vel Chnl	Froude #
Channel	Station	stationing	(cfs)	(ft)	Width, ft	(ft)	(ft)	(ft)	(ft/s)	Chl
Main Ditch	46	00+00	759	207.66	10.00	219.39	213.16	219.39	0.5	0.03
Main Ditch	45.5	00+30	759	207.66	10.00	219.39	213.16	219.39	0.5	0.03
Main Ditch	45.1	00+60	1,433	207.56	10.00	219.39	214.47	219.39	0.9	0.05
Main Ditch	45	01+00	1,433	207.41	10.00	219.39	213.84	219.39	0.9	0.05
Main Ditch	44	02+00	1,433	207.19	10.00	219.39	213.68	219.39	0.9	0.05
Main Ditch	43	03+00	1,433	206.98	10.00	219.39	213.54	219.39	0.9	0.05
Main Ditch	42	04+00	1,433	206.76	10.00	219.39		219.39	0.8	0.05
Main Ditch	41	05+00	1,433	206.54	10.00	219.39	212.99	219.39	0.9	0.05
Main Ditch	40	06+00	1,433	206.32	10.00	219.38	212.32	219.39	1.1	0.06
Main Ditch	39	07+00	1,433	206.11	10.00	219.38	212.09	219.39	0.9	0.05
Main Ditch	38	08+00	1,433	205.89	10.00	219.38	211.83	219.39	0.9	0.05
Main Ditch	37	09+00	1,433	205.67	10.00	219.38	211.58	219.39	0.9	0.05
Main Ditch	36	10+00	2,546	205.45	10.00	219.37	213.59	219.39	1.6	0.09
Main Ditch	35	11+00	2,546	205.24	10.00	219.37	213.31	219.38	1.6	0.09
Main Ditch	34	12+00	2,546	205.02	10.00	219.37	213.03	219.38	1.5	0.09
Main Ditch	33	13+00	2,546	204.80	10.00	219.36	212.28	219.38	1.6	0.08
Main Ditch	32	14+00	2,546	204.58	10.00	219.36	212.02	219.38	1.4	0.08
Main Ditch	31	15+00	2,546	204.37	10.00	219.36	211.75	219.38	1.5	0.08
Main Ditch	30	16+00	2,546	204.15	10.00	219.36	211.49	219.38	1.5	0.08
Main Ditch	29	17+00	2,546	203.93	10.00	219.35	211.22	219.38	1.7	0.09
Main Ditch	28	18+00	2,546	203.71	10.00	219.35	210.96	219.37	1.6	0.08
Main Ditch	27	19+00	2,546	203.50	10.00	219.35	210.69	219.37	1.6	0.08
Main Ditch	26.95	19+05	2,546	203.49	10.00	219.35	211.37	219.37	1.7	0.09
Main Ditch	26	20+00	6,291	203.28	14.00	219.26		219.36	3.7	0.19
Main Ditch	25	21+00	6,291	203.06	14.00	219.17		219.35	4.5	0.23
Main Ditch	24.44	21+56	6,291	202.94	14.00	219.17		219.34	4.4	0.22
Main Ditch	24.39	21+61	6,291	202.94	14.00	219.20		219.32	3.9	0.20
Main Ditch	24	22+00	6,367	202.84	14.00	218.90		219.29	6.0	0.32
Main Ditch	23	23+00	6,367	202.63	14.00	218.84		219.27	6.1	0.33
Main Ditch	22	24+00	6,367	202.41	14.00	218.42		219.21	7.6	0.42
Main Ditch	21	25+00	6,367	202.19	14.00	217.94	213.62	219.13	8.8	0.50
Main Ditch	20	26+00	6,367	201.97	14.00	217.92		219.08	8.7	0.49
Main Ditch	19	27+00	6,835	201.88	14.00	217.59		219.01	9.6	0.55
Main Ditch	18	28+00	6,835	201.54	14.00	217.61		218.93	9.2	0.53
Main Ditch	17	29+00	6,835	201.32	14.00	217.60		218.86	9.0	0.51
Main Ditch	16	30+00	6,835	201.00	14.00	217.67		218.78	8.5	0.47
Main Ditch	15	31+00	6,835	200.89	14.00	217.72		218.71	8.2	0.44
Main Ditch	14	32+00	6,835	200.67	14.00	217.80		218.63	7.6	0.41
Main Ditch	13	33+00	8,463	200.45	14.00	216.97		218.51	10.3	0.57
Main Ditch	12	34+00	8,463	200.23	14.00	216.54	213.37	218.42	11.0	0.63
Main Ditch	11	35+00	8,463	200.02	14.00	216.57		218.31	10.7	0.60
Main Ditch	10	36+00	8,463	199.80	14.00	216.54		218.23	10.5	0.59
Main Ditch	9	37+00	8,463	199.58	14.00	216.58		218.14	10.1	0.56
Main Ditch	8	38+00	9,919	199.36	14.00	213.50	213.50	217.77	16.6	1.00
Main Ditch	7	39+00	9,919	198.68	14.00	211.54	213.15	217.39	19.4	1.22
Main Ditch	6	40+00	9,919	198.00	14.00	209.75	211.96	216.92	21.7	1.49
Main Ditch	5	41+00	9,919	196.51	14.00	207.34	210.13	216.27	24.4	1.66
Main Ditch	4	42+00	9,919	192.00	14.00	198.94	202.49	214.63	36.3	3.15
Main Ditch	3	43+00	9,919	183.22	14.00	189.47	193.41	211.29	41.9	3.90
Main Ditch	2	44+00	9,919	172.00	14.00	179.37	179.37	181.90	17.3	1.24
Main Ditch	1	45+00	11,021	161.21	14.00	167.41	170.79	180.33	30.7	2.39

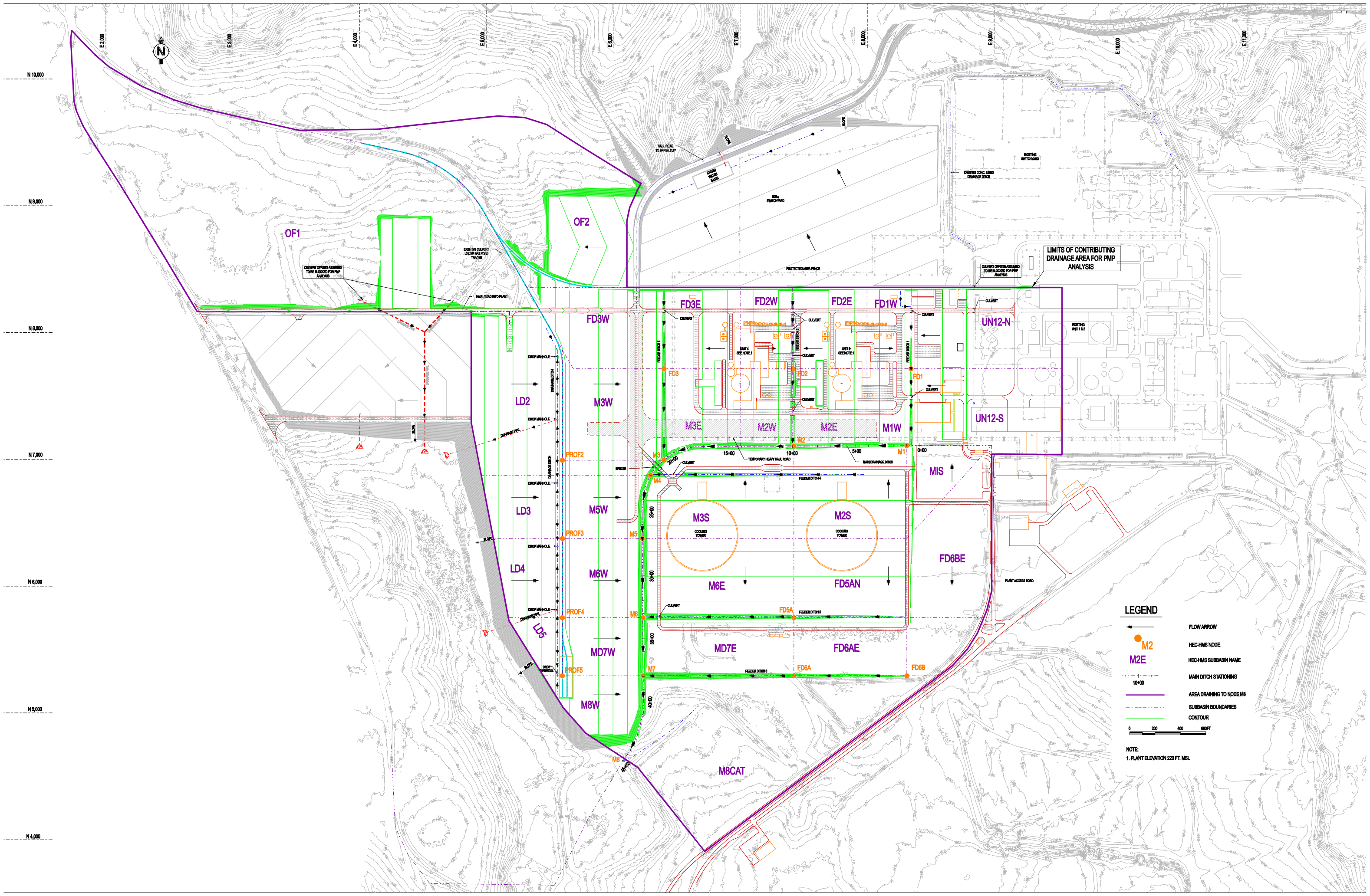
VEGP COL 2.4-2

Table 2.4-207 (Sheet 2 of 3)
Summary of HEC-RAS output for PMP Profile

HEC-RAS identification		Section	Q Total	Min Ch El	Bottom	W.S. Elev	Crit W.S.	E.G. Elev	Vel Chnl	Froude #
Channel	Station	stationing	(cfs)	(ft)	Width, ft	(ft)	(ft)	(ft)	(ft/s)	Chl
Feeder Ditch 3	83	83+00	568	211.70	5.00	219.43	214.83	219.44	0.4	0.02
Feeder Ditch 3	82	82+00	1,136	211.20	5.00	219.43		219.43	0.8	0.05
Feeder Ditch 3	81.9	81+90	1,136	211.15	5.00	219.43	215.22	219.43	0.7	0.05
Feeder Ditch 3	81.75	81+75	Inl Struct							
Feeder Ditch 3	81.6	81+60	1,136	211.00	5.00	219.42		219.42	0.7	0.04
Feeder Ditch 3	81	81+00	1,705	210.70	5.00	219.41		219.42	1.5	0.10
Feeder Ditch 3	80	80+00	2,273	210.20	5.00	219.40		219.42	2.0	0.12
Feeder Ditch 3	79	79+00	2,841	209.70	5.00	219.38		219.42	2.5	0.16
Feeder Ditch 3	78	78+00	3,409	209.20	5.00	219.36		219.41	3.0	0.19
Feeder Ditch 3	77	77+00	3,978	208.70	5.00	219.33		219.40	3.4	0.21
Feeder Ditch 3	76	76+00	4,172	208.20	5.00	219.34		219.39	3.0	0.18
Feeder Ditch 3	75	75+00	4,366	207.70	5.00	219.34		219.39	2.9	0.17
Feeder Ditch 3	74	74+00	4,561	207.20	5.00	219.32		219.38	3.1	0.18
Feeder Ditch 3	73	73+00	4,755	206.70	5.00	219.31		219.37	3.2	0.18
Feeder Ditch 3	72	72+00	4,949	206.20	5.00	219.30		219.37	3.3	0.19
Feeder Ditch 3	71	71+00	5,144	205.70	5.00	219.28		219.36	3.4	0.19
Feeder Ditch 2	83	83+00	62	214.26	5.00	219.42	214.99	219.42	0.1	0.00
Feeder Ditch 2	82	82+00	124	212.76	5.00	219.42		219.42	0.1	0.01
Feeder Ditch 2	81.95	81+95	124	212.74	5.00	219.42	214.85	219.42	0.1	0.01
Feeder Ditch 2	81.65	81+65	Inl Struct							
Feeder Ditch 2	81.35	81+35	124	212.44	5.00	219.41		219.41	0.1	0.01
Feeder Ditch 2	81	81+00	186	212.26	5.00	219.41		219.41	0.2	0.01
Feeder Ditch 2	80.65	80+65	186	212.06	5.00	219.41	214.65	219.41	0.2	0.01
Feeder Ditch 2	80.3	80+30	Inl Struct							
Feeder Ditch 2	80	80+00	249	211.76	5.00	219.41		219.41	0.2	0.02
Feeder Ditch 2	79.95	79+95	249	211.74	5.00	219.41		219.41	0.2	0.02
Feeder Ditch 2	79	79+00	311	211.26	5.00	219.41		219.41	0.3	0.02
Feeder Ditch 2	78.45	78+45	311	210.98	5.00	219.41	213.92	219.41	0.3	0.02
Feeder Ditch 2	78.2	78+20	Inl Struct							
Feeder Ditch 2	78	78+00	373	210.76	5.00	219.40		219.40	0.3	0.02
Feeder Ditch 2	77.95	77+95	373	210.74	5.00	219.40		219.40	0.3	0.02
Feeder Ditch 2	77	77+00	435	210.26	5.00	219.40		219.40	0.5	0.03
Feeder Ditch 2	76	76+00	535	209.76	5.00	219.40		219.40	0.8	0.05
Feeder Ditch 2	75	75+00	634	209.26	5.00	219.40		219.40	0.6	0.04
Feeder Ditch 2	74.95	74+95	634	209.24	5.00	219.40	213.35	219.40	0.6	0.04
Feeder Ditch 2	74.7	74+70	Inl Struct							
Feeder Ditch 2	74.45	74+45	634	208.99	5.00	219.39		219.39	0.6	0.04
Feeder Ditch 2	74.15	74+15	734	208.33	5.00	219.39	212.75	219.39	0.6	0.04
Feeder Ditch 2	74.1	74+10	Inl Struct							
Feeder Ditch 2	73.35	73+35	734	208.43	5.00	219.39		219.39	0.7	0.05
Feeder Ditch 2	73	73+00	834	208.26	5.00	219.38		219.39	0.8	0.05
Feeder Ditch 2	72	72+00	933	207.76	5.00	219.38		219.39	0.8	0.05
Feeder Ditch 2	71	71+00	1,033	207.26	5.00	219.38		219.39	0.9	0.06
Feeder Ditch 1	83	83+00	103	216.00	5.00	219.47	216.65	219.47	0.2	0.02
Feeder Ditch 1	82	82+00	206	215.70	5.00	219.47	216.85	219.47	0.4	0.03

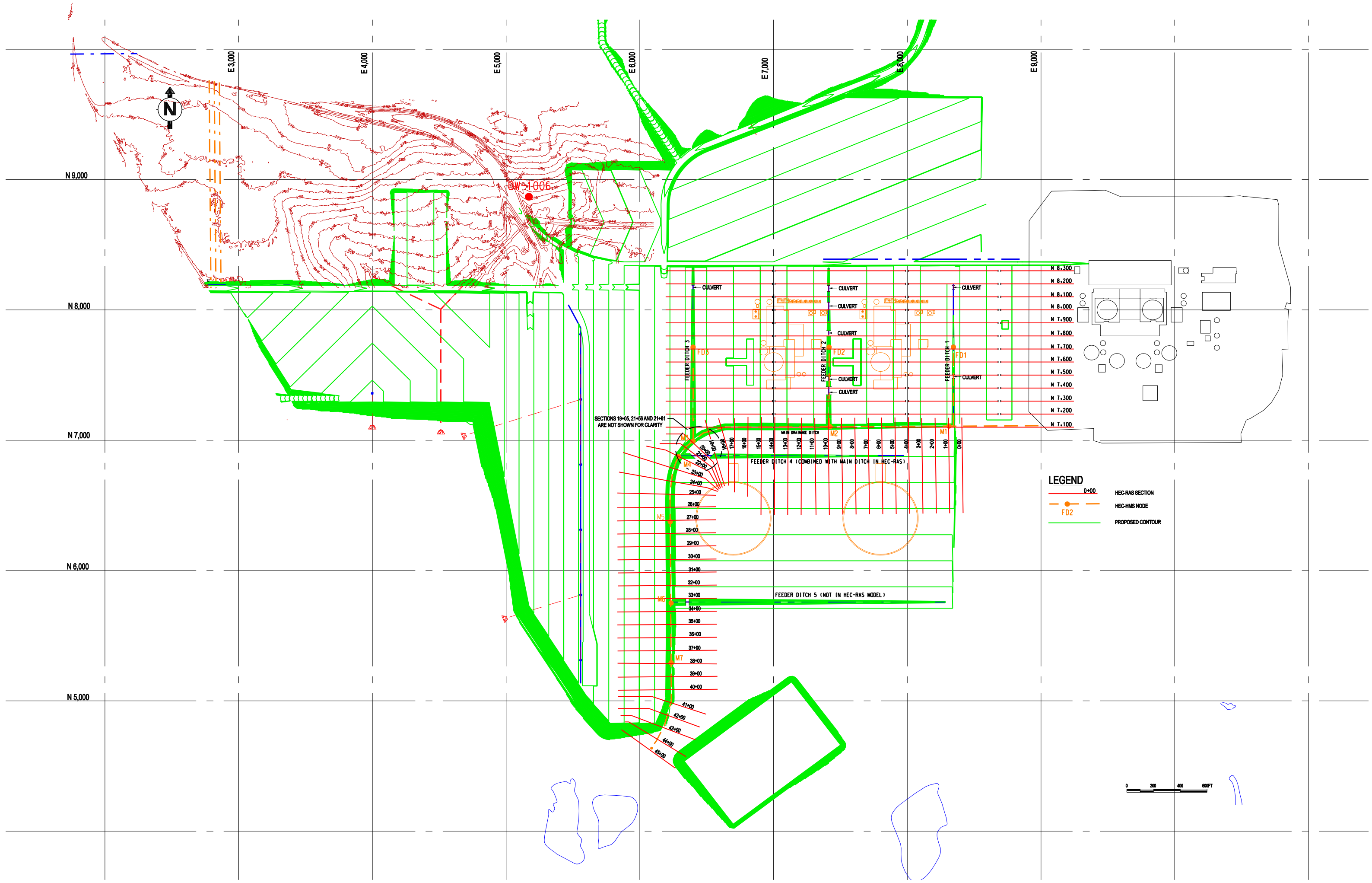
Table 2.4-207 (Sheet 3 of 3)
VEGP COL 2.4-2
Summary of HEC-RAS output for PMP Profile

HEC-RAS identification		Section	Q Total	Min Ch El	Bottom	W.S. Elev	Crit W.S.	E.G. Elev	Vel Chnl	Froude #
Channel	Station	stationing	(cfs)	(ft)	Width, ft	(ft)	(ft)	(ft)	(ft/s)	Chl
Feeder Ditch 1	81.9	81+90	206	215.65	5.00	219.47	216.81	219.47	0.4	0.03
Feeder Ditch 1	81.75	81+75	Inl Struct							
Feeder Ditch 1	81.6	81+60	206	215.50	5.00	219.46	216.67	219.46	0.4	0.03
Feeder Ditch 1	81	81+00	309	215.20	5.00	219.46	217.02	219.46	0.6	0.05
Feeder Ditch 1	80	80+00	413	214.70	5.00	219.46	217.15	219.46	0.8	0.07
Feeder Ditch 1	79	79+00	516	214.20	5.00	219.46	217.26	219.46	1.0	0.08
Feeder Ditch 1	78	78+00	619	213.70	5.00	219.45	217.35	219.46	1.2	0.10
Feeder Ditch 1	77	77+00	722	213.20	5.00	219.44	217.40	219.46	1.5	0.12
Feeder Ditch 1	76	76+00	843	212.70	5.00	219.43	217.43	219.45	1.8	0.14
Feeder Ditch 1	75.1	75+10	843	212.25	5.00	219.43	217.30	219.45	1.7	0.12
Feeder Ditch 1	75	75+00	964	212.20	5.00	219.43	217.41	219.45	1.9	0.14
Feeder Ditch 1	74.85	74+85	Inl Struct							
Feeder Ditch 1	74.7	74+70	964	212.05	5.00	219.42	217.26	219.44	1.7	0.13
Feeder Ditch 1	74	74+00	1,085	211.70	5.00	219.39	217.38	219.43	2.4	0.18
Feeder Ditch 1	73	73+00	1,205	211.20	5.00	219.38	217.29	219.42	2.3	0.17
Feeder Ditch 1	72	72+00	1,326	210.70	5.00	219.37	217.13	219.42	2.5	0.18
Feeder Ditch 1	71	71+00	1,447	210.20	5.00	219.35	216.22	219.41	2.7	0.18



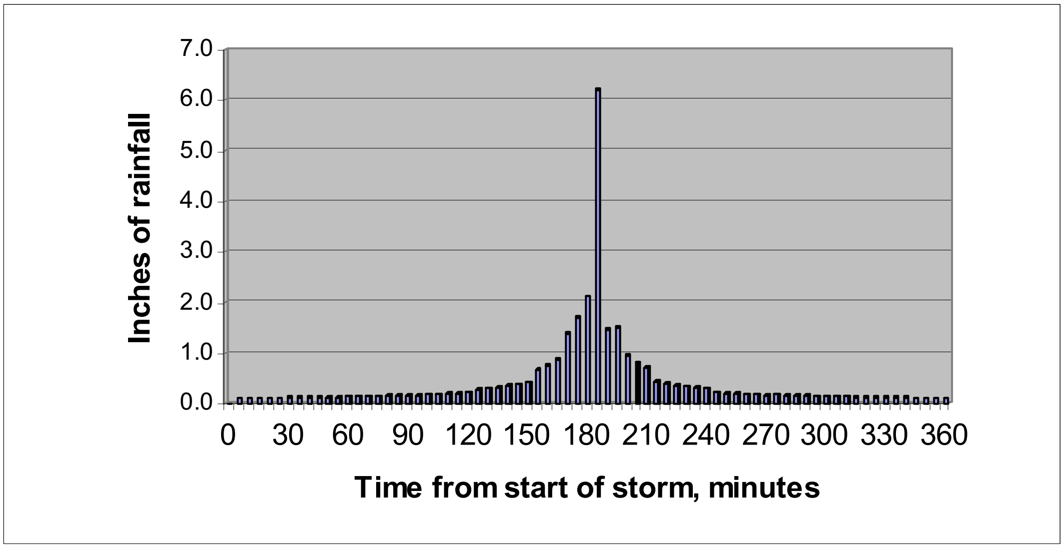
VEGP COL 2.4-2

Figure 2.4-201
Site Plan with PMP Drainage Boundaries and Flow Paths



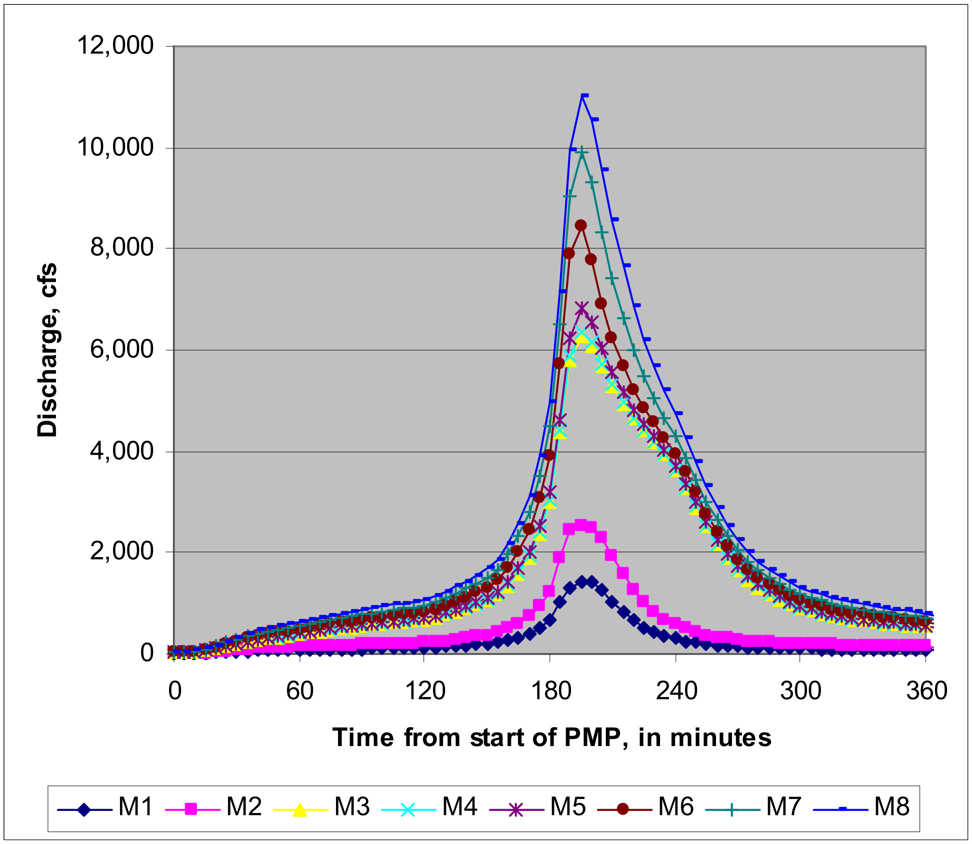
VEGP COL 2.4-2

Figure 2.4-201a
Cross-Section Location Map for HEC-RAS Model of Local PMF for Units 3 and 4



VEGP COL 2.4-2

Figure 2.4-202
PMP Hyetograph Determined in Frequency Storm Module of HEC-HMS



VEGP COL 2.4-2

Figure 2.4-203
HEC-HMS PMP Runoff Hydrographs at Points along Main Ditch

2.5 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

This section of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

2.5.4 STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

2.5.4.1.3 Mudmat

Replace DCD Subsection 2.5.4.1.3 with the folllowing text.

VEGP DEP 2.5-1 The mudmat provides a working surface prior to initiating the placement of reinforcement for the foundation mat structural concrete. The lower and upper mudmats are as follows:

- Lower mudmat — (6-inch layer) of un-reinforced concrete, with a minimum compressive strength of 2,500 psi. The lower mudmat will be used as the final dental concrete layer on the underlying foundation media.

- Upper mudmat — (6-inch layer) of un-reinforced concrete with a minimum compressive strength of 2,500 psi. This upper mudmat will support the chairs that, in turn, support the reinforcing steel.

The lower and upper mudmats are additionally described in ESPA SSAR Subsection 3.8.5.1.

The waterproofing system is described in DCD Subsection 2.5.4.6.12 and ESPA SSAR Subsection 3.8.5.1.1.

2.5.4.10 Static Stability

Insert the following new subsection after ESPA SSAR Subsection 2.5.4.10.2.

2.5.4.10.3 Lateral Earth Pressure

VEGP COL 2.5-11 The development of lateral earth pressures, static and dynamic (seismic), against the below-grade walls of safety-related structures is expected to be minimized with the construction of the mechanically stabilized earth (MSE) walls. As described in ESPA SSAR Subsection 2.5.4.5.7, the MSE walls are constructed adjacent to the Nuclear Island (NI) to facilitate the placement of backfill in the powerblock excavation. This bottom-up construction occurs prior to construction of the NI, and the MSE walls serve as the outside form for the NI below-grade walls. Although the MSE walls are expected to relieve much of the static lateral earth pressures exerted on the below-grade walls, over time these pressures may be transferred to the below-grade structure. Thus, the evaluation of site-specific lateral earth pressures for safety-related structures does not consider any influence from the MSE walls and full at-rest lateral earth pressures are assumed.

Site-specific static lateral earth pressures, assuming frictionless vertical walls and horizontally placed backfill, are evaluated using Rankine's theory for active, at-rest, and passive conditions (Reference 201). The earth pressure coefficients, k_a = 0.26, k_o, = 0.4, and k_p = 3.9 are based on a drained friction angle of 36 degrees for the compacted structural fill as presented in ESPA SSAR Table 2.5.4-1a. The at-rest earth pressure coefficient, k_o, for the compacted structural fill against the NI below grade walls is conservatively taken as 0.5.

The evaluation of site-specific lateral earth pressures includes the influence from surcharges. A vertical areal surcharge of 2,500 psf is used. This pressure conservatively represents construction loading prior to construction of adjacent buildings and subsequent adjacent permanent building loads. The vertical areal surcharge of 2,500 psf equates to a lateral surcharge pressure of 1,250 psf, which exceeds the AP1000 maximum lateral static plus dynamic design surcharge pressures.

Close-in compaction (behind the MSE wall) with a heavy vibratory roller is also considered. Lateral earth pressures increase as a result of compaction. These pressures are controlled at the construction stage by limiting the size of compaction equipment and its proximity to the walls. The influence of compaction was evaluated based on the characteristics of the vibratory compactor used for the Phase 1 Test Pad program (ESPA SSAR Appendix 2.5D). Compaction induced lateral earth pressures under at-rest conditions were evaluated using procedures developed by Duncan, et. al. (Reference 203). The inclusion of compaction-induced pressures is conservative given that these pressures will be exerted on the MSE wall prior to construction of the below-grade NI walls.

Site-specific seismic lateral earth pressures are evaluated for at-rest conditions using ASCE 4-98 (Reference 202). The site-specific ground acceleration at a frequency of 100 hertz for the Vogtle 3 and 4 site is taken as 0.266g (ESPA SSAR Subsection 2.5.2.6, Table 2.5.2-22b and Figure 2.5.2-38b).

Hydrostatic pressures, attributed to the groundwater level, exert lateral pressure on below-grade structures. At the VEGP Units 3 and 4 site, in the power block areas, the design groundwater elevation of 165 ft msl, as noted in ESPA SSAR Subsection 2.4.12, is about 15 feet below the NI basemat elevation of approximately 180 ft msl. The post construction water level, as identified in ESPA SSAR Appendix 2.4B, will also be well below the basemat elevation. Since the groundwater level is located well below the basemat, hydrostatic forces will not be exerted on the below-grade walls and hydrostatic pressures are not considered in the site-specific evaluation of lateral earth pressure for the NI.

In summary, Figure 2.5-201 presents the site-specific total at-rest lateral earth pressures for the below grade NI rigid walls. This diagram was developed assuming level ground surface, a post construction groundwater level below the basemat elevation (no hydrostatic pressure), an areal surcharge pressure of 2,500 psf, and compaction-induced pressure increases. Figure 2.5-202 presents the comparison of the site-specific total at-rest lateral earth pressure distribution compared to the AP1000 DCD design envelope in both the N-S and E-W directions. In both cases, the site-specific at-rest earth pressure is enveloped by the DCD design earth pressure envelopes by significant margins.

2.5.4.13 Heavy Lift Derrick Counterweight and Ring Foundation

---

VEGP SUP 2.5-1 The ring foundation for the heavy lift derrick (HLD) and counterweight are abandoned in place below grade following construction of Units 3 and 4. The HLD rails are removed from the ring foundation after construction of Units 3 and 4. The (HLD) counterweight and ring foundation are shown on Figure 2.5-203.

The top of the HLD counterweight and ring foundation concrete is located at approximately elevation 215 ft MSL, which is five feet below the nominal site grade of 220 ft MSL. The HLD counterweight and ring foundation are not visible following the installation of the roads, drainage provisions, and ground surface cover.

The HLD counterweight and ring foundation are below the surface drainage system provisions and do not affect the runoff for the local PMP flood event discussed in Subsection 2.4.2.3. The HLD counterweight and ring foundation are located above the design ground water elevation of 165 ft MSL, and do not impact the hydrological analyses described in ESPA SSAR Subsections 2.4.12 and 2.4.13.

The safety-related portion of the excavations is filled with Category 1 backfill to the NI basemat and with Category 2 backfill to grade. The side slopes are filled with engineered granular backfill (EGB), which is non-safety related and does not affect the static or seismic performance of the safety-related structures.

As shown on Figure 2.5-203, the HLD counterweight and ring foundation does not extend into the safety-related backfill of either Unit 3 or Unit 4. The ring foundation does extend into the EGB backfill of the excavations for both Unit 3 and Unit 4. The counterweight overall depth is approximately 28 ft. and is below the EGB backfill of the excavation for Unit 4. Subsection 3.7.1.1.1 provides the results of the evaluation which confirms that the presence of the HLD counterweight and ring foundation has no effect on the site specific seismic analyses.

VEGP DEP 1.1-1 This section is numbered in accordance with the referenced ESPA SSAR. The COL Information Items in DCD Subsections 2.5.1 through 2.5.6 are addressed in Subsection 2.5.7.

VEGP DEP 1.1-1

2.5.7 COMBINED LICENSE INFORMATION

2.5.7.1 Basic Geologic and Seismic Information

VEGP COL 2.5-1 This COL item is addressed in ESPA SSAR Subsections 2.5.1, 2.5.2, and 2.5.4.

2.5.7.2 Site Seismic and Tectonic Characteristics Information

VEGP COL 2.5-2 This COL item is addressed in ESPA SSAR Subsections 2.5.2 and 2.5.4.

2.5.7.3 Geoscience Parameters

VEGP COL 2.5-3 This COL item is addressed in ESPA SSAR Subsections 2.5.2 and 2.5.4.

2.5.7.4 Surface Faulting

VEGP COL 2.5-4 This COL item is addressed in ESPA SSAR Subsection 2.5.3.

2.5.7.5 Site and Structures

VEGP COL 2.5-5 This COL item is addressed in ESPA SSAR Subsection 2.5.4.

2.5.7.6 Properties of Underlying Materials

VEGP COL 2.5-6 This COL item is addressed in ESPA SSAR Subsection 2.5.4.

2.5.7.7 Excavation and Backfill

VEGP COL 2.5-7 This COL item is addressed in ESPA SSAR Subsection 2.5.4.

2.5.7.8 Groundwater Conditions

VEGP COL 2.5-8 This COL item is addressed in ESPA SSAR Subsection 2.5.4.

2.5.7.9 Liquefaction Potential

VEGP COL 2.5-9 This COL item is addressed in ESPA SSAR Subsection 2.5.4.

2.5.7.10 Bearing Capacity

VEGP COL 2.5-10 This COL item is addressed in ESPA SSAR Subsection 2.5.4.

2.5.7.11 Earth Pressures

VEGP COL 2.5-11 This COL item is addressed in ESPA SSAR Subsection 2.5.4.

2.5.7.12 Static and Dynamic Stability of Facilities

VEGP COL 2.5-12 This COL item is addressed in ESPA SSAR Subsection 2.5.4.

2.5.7.13 Subsurface Instrumentation

VEGP COL 2.5-13 This COL item is addressed in ESPA SSAR Section 2.5, Appendix 2.5E.

2.5.7.14 Stability of Slopes

VEGP COL 2.5-14 This COL item is addressed in ESPA SSAR Subsection 2.5.5.

2.5.7.15 Embankments and Dams

VEGP COL 2.5-15 This COL item is addressed in ESPA SSAR Subsection 2.5.6.

2.5.7.16 Settlement of Nuclear Island

VEGP COL 2.5-16 This COL item is addressed in ESPA SSAR Subsection 2.5.4.

2.5.7.17 Waterproofing System

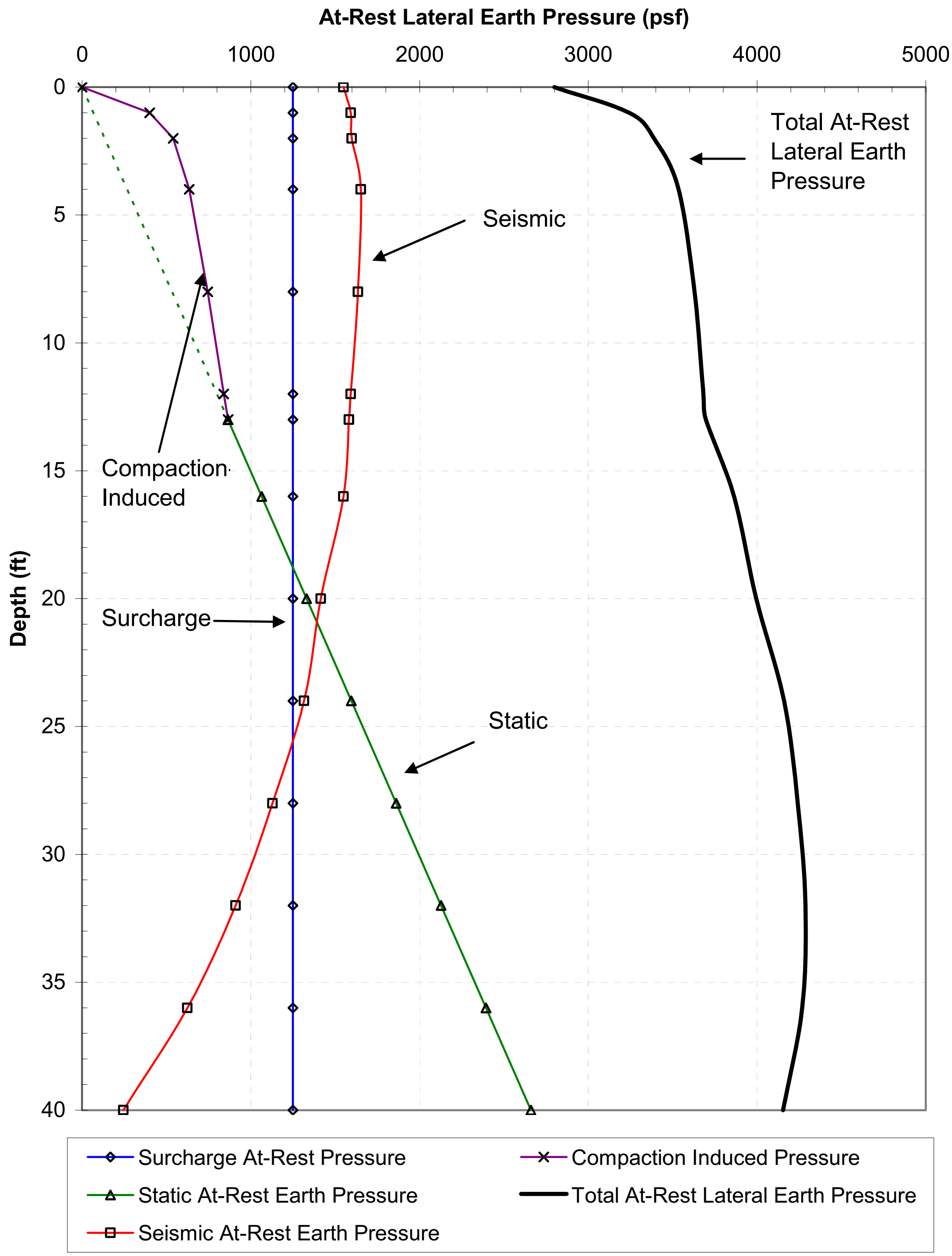
VEGP COL 2.5-17 This COL item is addressed in Subsections 3.8.5.1 and 3.4.1.1.1.1.

2.5.8 REFERENCES

201. Lambe, T.W. and R.V. Whitman, *Soil Mechanics*, John Wiley & Sons, Inc., New York, NY, 1969.

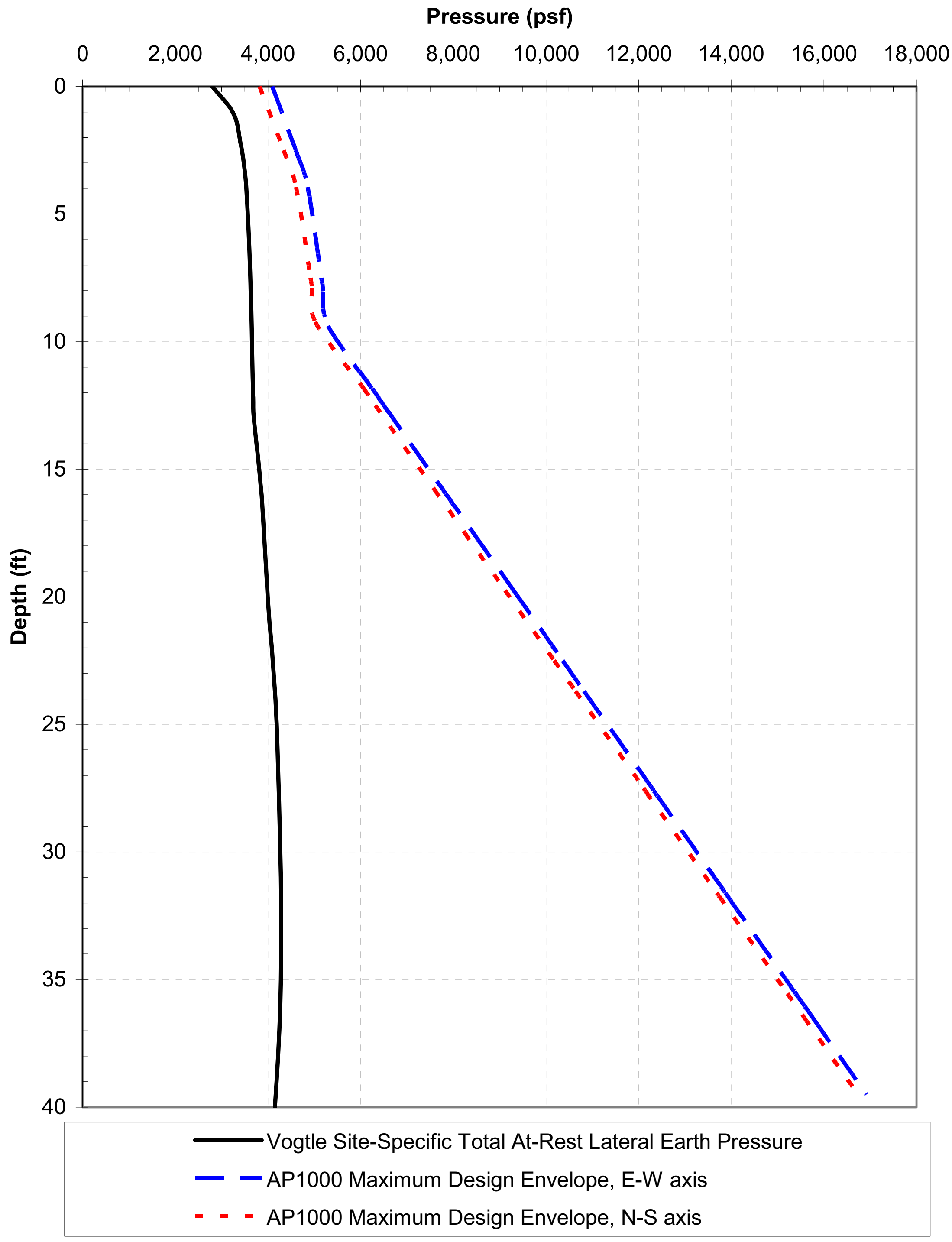
202. ASCE 4-98 (2000), *Seismic Analysis of Safety-Related Nuclear Structures and Commentary,* ASCE, Reston, VA, 2000.

203. Duncan, J.M., G.W. Williams, A.L. Sehn and R.B. Seed, “Closure of ‘Estimation of Earth Pressures due to Compaction’”, *Journal of Geotechnical Engineering*, ASCE, New York, NY, 119(7):1172-1177, July, 1993.



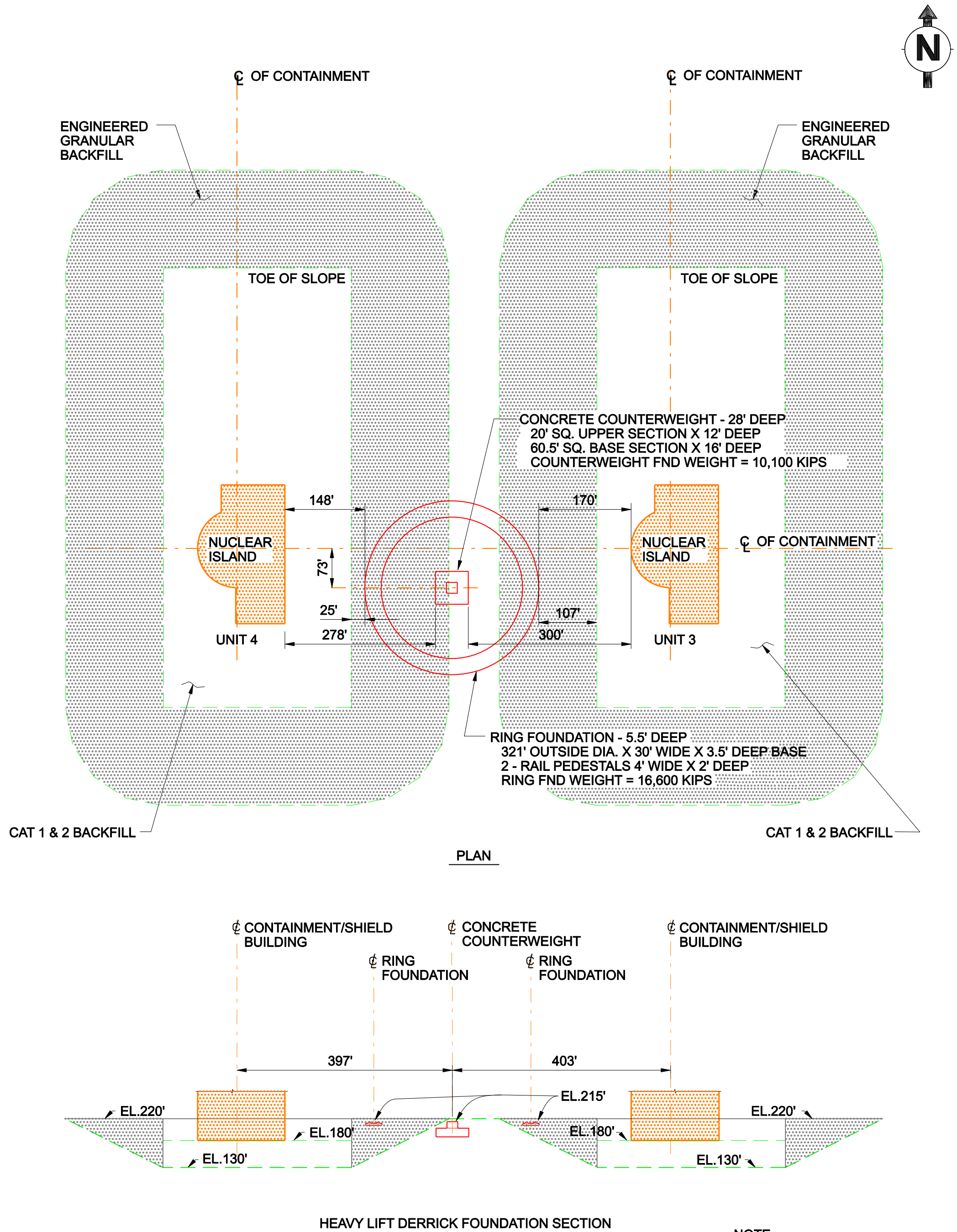
VEGP COL 2.5-11

Figure 2.5-201
Vogtle Site-Specific At-Rest Lateral Earth Pressure Diagrams for Rigid Below Grade Nuclear Island (NI) Walls



VEGP COL 2.5-11

Figure 2.5-202
Comparison of Vogtle Site-Specific Total At-Rest Lateral Earth Pressure Diagrams and AP1000 Maximum Design Envelopes



NOTE:
ALL DIMENSIONS, WEIGHTS, AND DISTANCES ARE APPROXIMATE

VEGP SUP 2.5-1

Figure 2.5-203
Heavy Lift Derrick Foundation Location

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

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CHAPTER 3
DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 CONFORMANCE WITH NUCLEAR REGULATORY COMMISSION GENERAL DESIGN CRITERIA

This section of the referenced DCD is incorporated by reference with no departures or supplements.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.2.1 SEISMIC CLASSIFICATION

Add the following text to the end of DCD Subsection 3.2.1.

VEGP SUP 3.2-1 There are no safety-related structures, systems, or components outside the scope of the DCD, except for engineered fill which is classified as a Seismic Category I, safety-related structure. See Table 3.2-201. Refer to ESPA SSAR Subsection 2.5.4 for a discussion of safety-related backfill.

The nonsafety-related structures, systems, and components outside the scope of the DCD are classified as non-seismic (NS).

3.2.2 AP1000 CLASSIFICATION SYSTEM

Add the following text to the end of DCD Subsection 3.2.2.

VEGP SUP 3.2-1 There are no safety-related structures, systems, or components outside the scope of the DCD, except for engineered fill which is classified as a Seismic Category I, safety-related structure. See Table 3.2-201. Refer to ESPA SSAR Subsection 2.5.4 for a discussion of safety-related backfill.

Table 3.2-201
VEGP SUP 3.2-1
Seismic Classification of Building Structures

Structure	Category
Safety-Related Backfill	C-I

C-I: Seismic Category I

3.3 WIND AND TORNADO LOADINGS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.3.1.1 Design Wind Velocity

Add the following text to the end of DCD Subsection 3.3.1.1.

VEGP COL 3.3-1
VEGP COL 3.5-1

The wind velocity characteristics for the Vogtle Electric Generating Plant, Units 3 and 4 (VEGP), are given in ESPA SSAR Subsection 2.3.1.3.1. These values are bounded by the design wind velocity values given in DCD Subsection 3.3.1.1 for the AP1000 plant.

3.3.2.1 Applicable Design Parameters

Add the following text to the end of DCD Subsection 3.3.2.1.

VEGP COL 3.3-1
VEGP COL 3.5-1

The tornado characteristics for the VEGP are given in ESPA SSAR Subsection 2.3.1.3.2. These values are bounded by the tornado design parameters given in DCD Subsection 3.3.2.1 for the AP1000 plant.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

---

Add the following text to the end of DCD Subsection 3.3.2.3.

VEGP COL 3.3-1
VEGP COL 3.5-1

Consideration of the effects of wind and tornado due to failures in an adjacent AP1000 plant and VEGP Units 1 and 2 are bounded by the evaluation of the buildings and structures in a single unit.

3.3.3 COMBINED LICENSE INFORMATION

Add the following text to the end of DCD Subsection 3.3.3.

VEGP COL 3.3-1 The VEGP site satisfies the site interface criteria for wind and tornado (see Subsections 3.3.1.1, 3.3.2.1 and 3.3.2.3) and will not have a tornado-initiated failure of structures and components within the applicant's scope that compromises the safety of AP1000 safety-related structures and components (see also Subsection 3.5.4).

Subsection 1.2.2 discusses differences between the plant specific site plan (see Figure 1.1-202) and the AP1000 typical site plan shown in DCD Figure 1.2-2.

There are no other structures adjacent to the nuclear island other than as described and evaluated in the DCD.

Missiles caused by external events separate from the tornado are addressed in Subsections 3.5.1.3, 3.5.1.5, and 3.5.1.6.

3.4 WATER LEVEL (FLOOD) DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.4.1.1.1.1 Waterproofing

Add the following text to the end of the fourth bullet of the first paragraph of DCD Subsection 3.4.1.1.1.1.

VEGP DEP 3.4-1
VEGP COL 2.5-17

An alternate waterproofing system for the seismic Category I structures below grade is as presented in ESPA SSAR Subsection 3.8.5.1.1.

3.4.1.3 Permanent Dewatering System

Add the following text to the end of DCD Subsection 3.4.1.3.

VEGP COL 3.4-1

No permanent dewatering system is required because site groundwater levels are two feet or more below site grade level as described in ESPA SSAR Subsection 2.4.12.

3.4.3 COMBINED LICENSE INFORMATION

Replace the first paragraph of DCD Subsection 3.4.3 with the following text.

VEGP COL 3.4-1

The site-specific water levels given in Subsection 3.4.1.3 and ESPA SSAR Subsection 2.4 satisfy the interface requirements identified in DCD Section 2.4.

3.5 MISSILE PROTECTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.5.1.3 Turbine Missiles

Add the following text to the end of DCD Subsection 3.5.1.3.

STD SUP 3.5-1 The potential for a turbine missile from another AP1000 plant in close proximity has been considered. As noted in DCD Subsection 10.2.2, the probability of generation of a turbine missile (or P1 as identified in SRP 3.5.1.3) is less than 1 x 10^{-5} per year. This missile generation probability (P1) combined with an unfavorable orientation P2 x P3 conservative product value of 10^{-2} (from SRP 3.5.1.3) results in a probability of unacceptable damage from turbine missiles (or P4 value) of less than 10^{-7} per year per plant which meets the SRP 3.5.1.3 acceptance criterion and the guidance of Regulatory Guide 1.115. Thus, neither the orientation of the side-by-side AP1000 turbines nor the separation distance is pertinent to meeting the turbine missile generation acceptance criterion. In addition, the shield building and auxiliary building walls, roofs, and floors, provide further conservative, inherent protection of the safety-related SSCs from a turbine missile.

VEGP SUP 3.5-1 The orientation of the Units 1 and 2 turbines has been evaluated and Vogtle Units 3 and 4 are located outside of the low trajectory strike zones as described in Regulatory Guide 1.115. Therefore, there is no potential for a turbine missile from Units 1 and 2 to impact Units 3 and 4.

STD SUP 3.5-2 The turbine system maintenance and inspection program is discussed in Subsection 10.2.3.6.

3.5.1.5 Missiles Generated by Events Near the Site

Add the following text to the end of DCD Subsection 3.5.1.5.

VEGP COL 3.3-1
VEGP COL 3.5-1 The primary access point, administrative building, communications support center, warehouse and shops, engineering and administrative building,

maintenance support building and miscellaneous structures 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. Additionally, there are no other structures adjacent to the nuclear island other than the turbine building, annex building, radwaste building and passive containment cooling ancillary water storage tank.

In accordance with ESPA SSAR Subsection 2.2.3, the effects of explosions have been evaluated and it has been determined that the overpressure criteria of Regulatory Guide 1.91 is not exceeded. Consistent with Regulatory Guide 1.91, the effects of blast-generated missiles will be less than those associated with the blast overpressure levels considered; therefore, no further evaluation of blast-generated missiles is required.

3.5.1.6 Aircraft Hazards

VEGP COL 3.3-1
VEGP COL 3.5-1

This section of the referenced ESPA SSAR is incorporated by reference with no variances or supplements.

3.5.4 COMBINED LICENSE INFORMATION

VEGP COL 3.5-1

Add the following text to the end of DCD Subsection 3.5.4.

The VEGP site satisfies the site interface criteria for wind and tornado (see Subsections 3.3.1.1, 3.3.2.1 and 3.3.2.3) and will not have a tornado-initiated failure of structures and components within the applicant's scope that compromises the safety of AP1000 safety-related structures and components (see also Subsection 3.3.3).

Subsection 1.2.2 discusses differences between the plant specific site plan (see Figure 1.1-202) and the AP1000 typical site plan shown in DCD Figure 1.2-2.

There are no other structures adjacent to the nuclear island other than as described and evaluated in the DCD.

Missiles caused by external events separate from the tornado are addressed in Subsections 3.5.1.3, 3.5.1.5, and 3.5.1.6.

3.6 PROTECTION AGAINST THE DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.6.4.1 Pipe Break Hazard Analysis

Replace the last paragraph in DCD Subsection 3.6.4.1 with the following text.

STD COL 3.6-1 The as-designed pipe rupture hazards evaluation is made available for NRC review. The completed as-designed pipe rupture hazards evaluation will be in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. Systems, structures, and components identified to be essential targets protected by associated mitigation features (Reference is DCD Table 3.6-3) will be confirmed as part of the evaluation, and updated information will be provided as appropriate.

A pipe rupture hazard analysis is part of the piping design. The evaluation will be performed for high and moderate energy piping to confirm the protection of systems, structures, and components which are required to be functional during and following a design basis event. The locations of the postulated ruptures and essential targets will be established and required pipe whip restraints and jet shield designs will be included. The report will address environmental and flooding effects of cracks in high and moderate energy piping. The as-designed pipe rupture hazards evaluation is prepared on a generic basis to address COL applications referencing the AP1000 design.

The pipe whip restraint and jet shield design includes the properties and characteristics of procured components connected to the piping, components, and walls at identified break and target locations. The design will be completed prior to installation of the piping and connected components.

The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5 will be completed prior to fuel load (in accordance with DCD Tier 1 Table 3.3-6, item 8).

This COL item is also addressed in Subsection 14.3.3.

3.6.4.4 Primary System Inspection Program for Leak-before-Break Piping

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Replace the first paragraph of DCD Subsection 3.6.4.4 with the following text.

STD COL 3.6-4 Alloy 690 is not used in leak-before-break piping. No additional or augmented inspections are required beyond the inservice inspection program for leak-before-break 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.

3.7 SEISMIC DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add Subsection 3.7.1.1.1 as follows:

VEGP SUP 3.7-3 3.7.1.1.1 Design Ground Motion Response Spectra

The Vogtle site-specific safe shutdown earthquake (SSE) design response spectra (DRS) are the site-specific ground motion response spectra (GMRS) determined in ESPA SSAR Subsection 2.5.2.6. These response spectra are determined in the free-field on the ground surface.

The Vogtle foundation input response spectra (FIRS) are at an outcrop located at the 40' depth. The development of these FIRS is discussed in ESPA SSAR Subsection 2.5.2.7. These Vogtle response spectra are compared to the AP1000 SSE design response spectra that are also referred to as the AP1000 certified seismic design response spectra (CSDRS). The CSDRS also represents the AP1000 FIRS. This is because: (1) the CSDRS at a hard rock site is essentially the same at grade and at foundation; and (2) the CSDRS envelopes the in-column motions of the other generic soil conditions. The AP1000 CSDRS are applied at the foundation level in the free field at hard rock sites, and at the finished grade for the other soil generic conditions. The comparisons are shown in Figures 3.7-201 and 3.7-202. As seen from those comparisons, there are exceedances above the CSDRS; therefore, plant specific seismic evaluations are performed that demonstrate that the AP1000 plant designed for the CSDRS is acceptable for the Vogtle site. The results from a Vogtle site specific two-dimensional seismic evaluation that demonstrates the acceptability of the Vogtle site are given in ESPA SSAR Appendix 2.5E. Additionally, a Vogtle site specific three-dimensional seismic evaluation that demonstrates the acceptability of the Vogtle site is given in Appendix 3GG. Based on these Vogtle site specific seismic evaluations it can also be concluded that the standard AP1000 plant certified design is fully acceptable to a SSE design response spectra level of the CSDRS at Vogtle's plant grade.

VEGP SUP 2.5-1 As discussed in Subsection 2.5.4.13, the heavy lift derrick (HLD) counterweight and ring foundation were abandoned in place after construction. The HLD counterweight is outside the defined excavation of Unit 3 and Unit 4 and therefore does not need to be evaluated. Portions of the HLD ring foundation extend over the Unit 3 and Unit 4 excavation slopes within the engineered granular backfill (EGB); but outside the Category 1 and 2 backfill. The presence of the HLD ring foundation has no effect on the VEGP site-specific 3D SASSI SSI analyses of the Nuclear Island (NI) presented in Appendix 3GG based on the following information.

The VEGP site-specific 3D SASSI SSI of the NI is consistent with the accepted DCD 3D SASSI NI modeling approach of not including structure-to-structure interaction of the adjacent structures such as the Annex Building and the Turbine Building; and therefore the more distant abandoned HLD ring foundation has even less structure-to-structure effects on the NI seismic response. Additionally only a portion of the abandoned HLD ring foundation is within a limited area of the non-safety EGB over the slopes of the excavation. It has been demonstrated in the ESP as amended that a large variation of the EGB properties does not significantly affect the site-specific seismic analyses; therefore, it is concluded the abandoned portion of the HLD ring foundation in the EGB has no significant effect on the site-specific seismic analyses.

VEGP SUP 3.7-3 The operating basis earthquake ground motion (OBE) spectral values are used as one measure of potential damage to those structures, systems, and components designed to the SSE design ground motion to determine the severity of the seismic event and make a determination of whether the plant must be shut down. For the AP1000 certified design, OBE is not an explicit design load; as such it is therefore defined as one-third the CSDRS. Since it has been demonstrated that the Vogtle site characteristics do not limit the AP1000 design to the CSDRS, the Vogtle OBE for the AP1000 is defined as one-third the AP1000 CSDRS.

The FIRS and the CSDRS in the horizontal direction in the free-field at the foundation of the AP1000 Nuclear Island exceed the minimum spectrum requirements of 10 CFR50 Appendix S.

3.7.2.12 Methods for Seismic Analysis of Dams

Add the following text to the end of DCD Subsection 3.7.2.12.

VEGP COL 3.7-1 The evaluation of existing dams whose failure could affect the site interface flood level specified in DCD Subsection 2.4.1.2, is included in ESPA SSAR Subsection 2.4.4. As discussed in ESPA SSAR Subsection 2.4.1.2.4, the U.S. Army Corps of Engineers has no current plans for the construction of additional reservoirs on the Savannah River.

3.7.4.1 Comparison with Regulatory Guide 1.12

Add the following text to the end of DCD Subsection 3.7.4.1.

STD SUP 3.7-1 Administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments in-service during plant operation and shutdown in accordance with Regulatory Guide 1.12.

3.7.4.2.1 Triaxial Acceleration Sensors

Add the following text to the end of DCD Subsection 3.7.4.2.1.

VEGP COL 3.7-5 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.

3.7.4.4 Comparison of Measured and Predicted Responses

Add the following text to the end of DCD Subsection 3.7.4.4.

VEGP COL 3.7-2 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 and 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.

STD COL 3.7-2 In addition, the procedures address 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, and provide for appropriate corrective actions to be taken if needed (such as repositioning the racks or analysis of the as-found condition).

3.7.4.5 Tests and Inspections

Add the following text to the end of DCD Subsection 3.7.4.5.

STD SUP 3.7-2 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.

3.7.5 COMBINED LICENSE INFORMATION

3.7.5.1 Seismic Analysis of Dams

VEGP COL 3.7-1 This COL Item is addressed in Subsection 3.7.2.12 and ESPA SSAR Subsection 2.4.4.

3.7.5.2 Post-Earthquake Procedures

VEGP COL 3.7-2
STD COL 3.7-2 This COL Item is addressed in Subsection 3.7.4.4.

3.7.5.3 Seismic Interaction Review

Replace DCD Subsection 3.7.5.3 with the following text.

STD COL 3.7-3 The seismic interaction review will be updated 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 completed prior to fuel load.

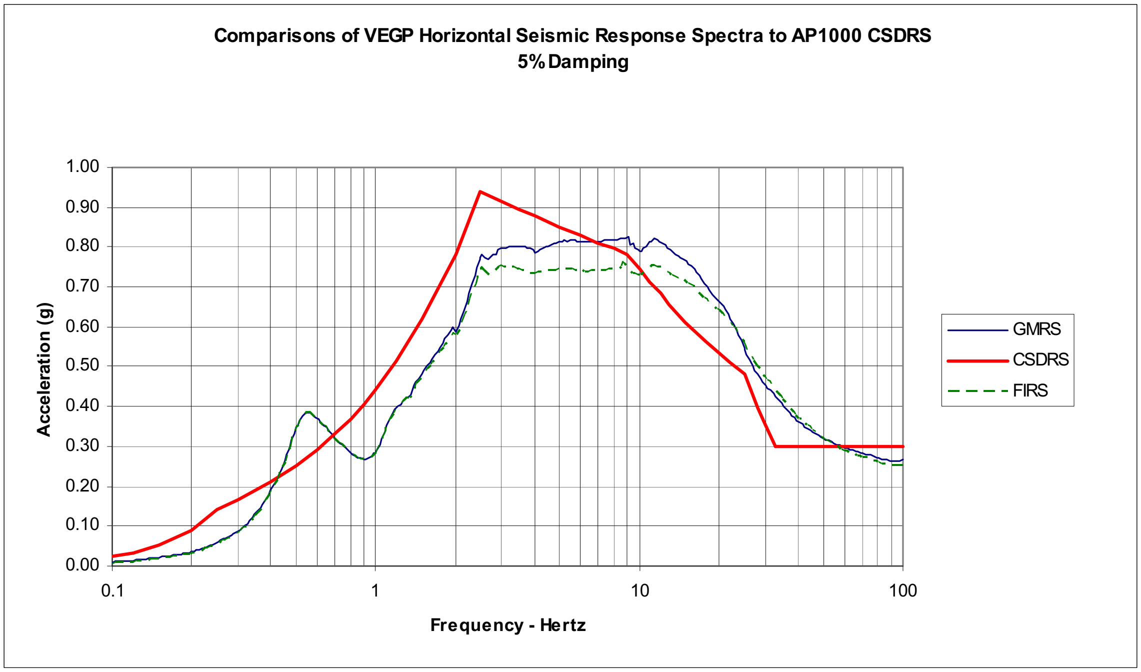
3.7.5.4 Reconciliation of Seismic Analyses of Nuclear Island Structures

Replace DCD Subsection 3.7.5.4 with the following text.

STD COL 3.7-4 The seismic analyses described in DCD Subsection 3.7.2 will be reconciled for detailed 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. Deviations are acceptable based on an evaluation consistent with the methods and procedure of DCD Section 3.7 provided the amplitude of the seismic floor response spectra, including the effect due to these deviations, does not exceed the design basis floor response spectra by more than 10 percent. This reconciliation will be completed prior to fuel load.

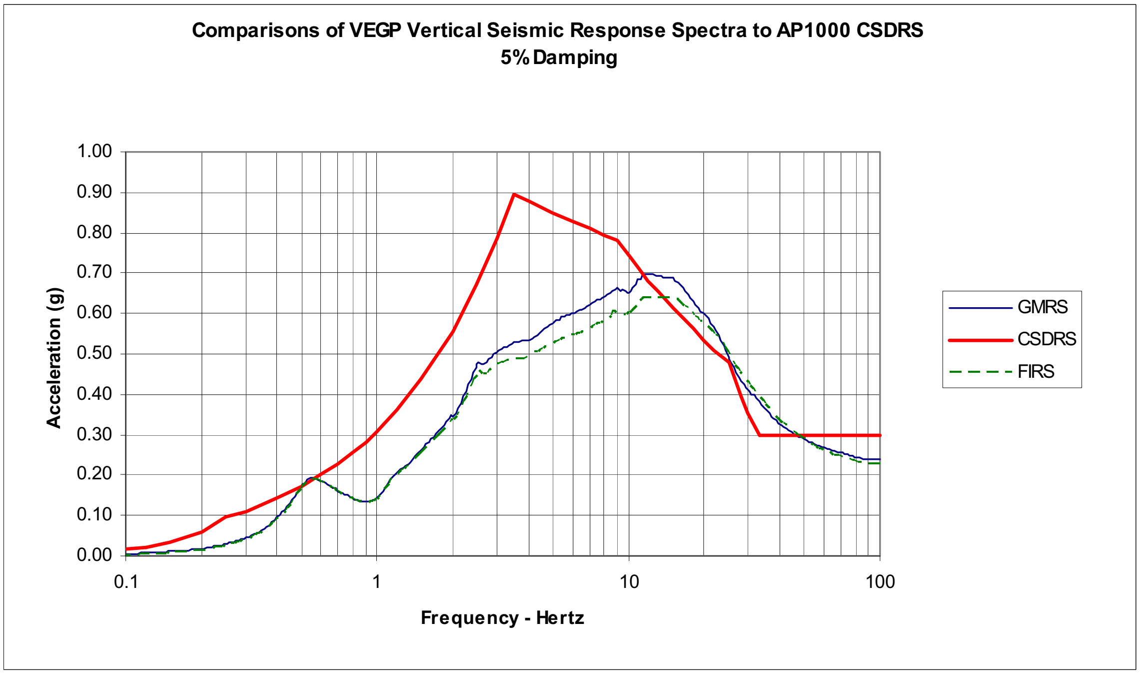
3.7.5.5 Free Field Acceleration Sensor

VEGP COL 3.7-5 This COL Item is addressed in Subsection 3.7.4.2.1.



VEGP SUP 3.7-3

Figure 3.7-201
VEGP AP1000 Horizontal Spectra Comparison



VEGP SUP 3.7-3

Figure 3.7-202
VEGP AP1000 Vertical Spectra Comparison

3.8 DESIGN OF CATEGORY I STRUCTURES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.8.3.7 In-Service Testing and Inspection Requirements

Replace the existing DCD statement with the following:

STD COL 3.8-5 The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.4.3 Loads and Load Combinations

Add the following to the end of Subsection 3.8.4.3.1.3:

VEGP SUP 3.8-2 The application of the 48-hour PMWP and the 100-year return period ground-level snowpack in the roof design of safety-related structures is addressed in ESPA SSAR Subsection 2.3.1.3.4.

3.8.4.7 Testing and In-Service Inspection Requirements

Replace the existing DCD final statement of the subsection with the following:

STD COL 3.8-5 The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.5.1 Description of the Foundations

Add the following text after paragraph one of DCD Subsection 3.8.5.1.

VEGP SUP 3.8-1 The depth of overburden and depth of embedment are given in ESPA SSAR Subsection 2.5.4.5.

VEGP SUP 3.8-3 A description of the safety-related backfill, which supports Category I structures, is given in ESPA SSAR Subsection 2.5.4.5.

VEGP ESP VAR 1.6-2
VEGP COL 2.5-17 Subsection 3.8.5 of the referenced ESPA SSAR is incorporated by reference after the last paragraph of DCD Subsection 3.8.5.1 with the following variance:

The first paragraph of ESPA SSAR Subsection 3.8.5.1, which pertains to DCD Revision 15, is not incorporated by reference.

In addition, the first paragraph in ESPA SSAR Subsection 3.8.5.1.1 also addresses material specific to Revision 15 of the DCD. Therefore, that paragraph is not incorporated by reference.

3.8.5.7 In-Service Testing and Inspection Requirements

Replace the existing DCD first statement with the following:

STD COL 3.8-5 The inspection program for structures is identified in Section 17.6. This inspection program is consistent with the requirements of 10 CFR 50.65 and the guidance in Regulatory Guide 1.160.

3.8.6.5 Structures Inspection Program

STD COL 3.8-5 This item is addressed in Subsections 3.8.3.7, 3.8.4.7, 3.8.5.7, and 17.6.

3.8.6.6 Construction Procedures Program

Add the following to the end of Subsection 3.8.6.6:

STD COL 3.8-6 Construction and inspection procedures for concrete filled steel plate modules address 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 procedures will be made available to NRC inspectors prior to use.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.9.3.1.2 Loads for Class 1 Components, Core Support, and Component Supports

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STD COL 3.9-5 Add the following after the last paragraph under DCD subheading Request 3) and prior to DCD subheading Other Applications.

PRESSURIZER SURGE LINE MONITORING

General

The pressurizer surge line is monitored at the first AP1000 plant to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters. This monitoring occurs during the hot functional testing and first fuel cycle. The resulting monitoring data is evaluated to verify that the pressurizer surge line is within the bounds of the analytical temperature distributions and displacements.

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.

The pressurizer surge line monitoring activities include the following methodology and requirements:

Monitoring Method

The pressurizer surge line pipe wall is instrumented with outside mounted temperature and displacement sensors. The data from this instrumentation is supplemented by plant computer data from related process and control parameters.

Locations to be Monitored

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.

Data Evaluation

Data evaluation is performed at the completion of the monitoring period (one fuel cycle). The evaluation includes a comparison of the data evaluation results with the thermal profiles and transient loadings defined for the pressurizer surge line, accounting for expected pipe outside wall temperatures. Interim evaluations of the data are performed during the hot functional testing period, up to the start of normal power operation, and again once three months worth of normal operating data has been collected, to identify any unexpected conditions in the pressurizer surge line.

3.9.3.4.4 Inspection, Testing, Repair, and/or Replacement of Snubbers

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Add the following text after the last paragraph of DCD Subsection 3.9.3.4.4:

STD SUP 3.9-3

a. Snubber Design and Testing

   1. A list of snubbers on systems which experience sufficient thermal movement to measure cold to hot position is included in Table 3.9-201.

   2. The snubbers are tested to verify they can perform as required during the seismic events, and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. Production and qualification test programs for both hydraulic and mechanical snubbers are carried out by the snubber vendors in accordance with the snubber installation instruction manual required to be furnished by the snubber supplier. Acceptance criteria for compliance with ASME Section III Subsection NF, and other applicable codes, standards, and requirements, are as follows:

      - Snubber production and qualification test programs are carried out by strict adherence to the manufacturer's snubber installation and instruction manual. This manual is prepared by the snubber manufacturer and subjected to review for compliance with the applicable provisions of the ASME Pressure Vessel and Piping Code of record. The test program is periodically audited during implementation for compliance.

- Snubbers are inspected and tested for compliance with the design drawings and functional requirements of the procurement specifications.

- Snubbers are inspected and qualification tested. No sampling methods are used in the qualification tests.

- Snubbers are load rated by testing in accordance with the snubber manufacturer's testing program and in compliance with the applicable sections of ASME QME-1-2007, Subsection QDR and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Subsection ISTD.

- Design compliance of the snubbers per ASME Section III Paragraph NF-3128, and Subparagraphs NF-3411.3 and NF-3412.4.

- The snubbers are tested for various abnormal environmental conditions. Upon completion of the abnormal environmental transient test, the snubber is tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test. The functional parameters cited in Subparagraph NF-3412.4 are included in the snubber qualification and testing program. Other parameters in accordance with applicable ASME QME-1-2007 and the ASME OM Code will be incorporated.

- The codes and standards used for snubber qualification and production testing are as follows:

  - ASME B&PV Code Section III (Code of Record date) and Subsection NF.

  - ASME QME-1-2007, Subsection QDR and ASME OM Code, Subsection ISTD.

- Large bore hydraulic snubbers are full Service Level D load tested, including verifying bleed rates, control valve closure within the specified velocity ranges and drag forces/breakaway forces are acceptable in accordance with ASME, QME-1-2007 and ASME OM Codes.

3. Safety-related snubbers are identified in Table 3.9-201, including the snubber identification and the associated system or component, e.g., line number. The snubbers on the list are hydraulic and constructed to ASME Section III, Subsection NF. The

snubbers are used for shock loading only. None of the snubbers are dual purpose or vibration arrestor type snubbers.

b. Snubber Installation Requirements

Installation instructions contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing that contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.

STD COL 3.9-3 The description of the snubber preservice and inservice testing programs in this section is based on the ASME OM Code 2001 Edition through 2003 Addenda. The initial inservice testing program incorporates 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. Limitations and modifications set forth in 10 CFR 50.55a are incorporated.

c. Snubber Preservice Examination and Testing

The preservice examination plan for applicable snubbers is prepared in accordance with the requirements of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Subsection ISTD, and the additional requirements of this Section. This examination is made after snubber installation but not more than 6 months prior to initial system preoperational testing. The preservice examination verifies the following:

1. There are no visible signs of damage or impaired operational readiness as a result of storage, handling, or installation.

2. The snubber load rating, location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

3. Snubbers are not seized, frozen or jammed.

4. Adequate swing clearance is provided to allow snubber movements.

5. If applicable, fluid is to the recommended level and is not to be leaking from the snubber system.

6. Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial preservice examination and initial system preoperational tests exceeds 6 months, reexamination of Items 1, 4, and 5 is performed. Snubbers, which are installed incorrectly or otherwise fail to meet the above requirements, are repaired or replaced and re-examined in accordance with the above criteria.

A preservice thermal movement examination is also performed, during initial system heatup and cooldown. For systems whose design operating temperature exceeds 250ºF (121ºC), snubber thermal movement is verified.

Additionally, preservice operational readiness testing is performed on snubbers. The operational readiness test is performed to verify the parameters of ISTD 5120. Snubbers that fail the preservice operational readiness test are evaluated to determine the cause of failure, and are retested following completion of corrective action(s).

Snubbers that are installed incorrectly or otherwise fail preservice testing requirements are re-installed correctly, adjusted, modified, repaired or replaced, as required. Preservice examination and testing is re-performed on installation-corrected, adjusted, modified, repaired or replaced snubbers as required.

d. Snubber Inservice Examination and Testing

Inservice examination and testing of safety-related snubbers is conducted in accordance with the requirements of the ASME OM Code, Subsection ISTD. Inservice examination is initially performed not less than two months after attaining 5 percent reactor power operation and is completed within 12 calendar months after attaining 5 percent reactor power. Subsequent examinations are performed at intervals defined by ISTD-4252 and Table ISTD-4252-1. Examination intervals, subsequent to the third interval, are adjusted based on the number of unacceptable snubbers identified in the current interval.

An inservice visual examination is performed on the snubbers to identify physical damage, leakage, corrosion, degradation, indication of binding, misalignment or deformation and potential defects generic to a particular design. Snubbers that do not meet visual examination requirements are evaluated to determine the root cause of the unacceptability, and appropriate corrective actions (e.g., snubber is adjusted, repaired, modified, or replaced) are taken. Snubbers evaluated as unacceptable during visual examination may be accepted for continued service by successful completion of an operational readiness test.

Snubbers are tested inservice to determine operational readiness during each fuel cycle, beginning no sooner than 60 days before the start of the refueling outage. Snubber operational readiness tests are conducted with the snubber in the as-found condition, to the extent practical, either in-place or on a test bench, to verify the test parameters of ISTD-5210. When an in-place test or bench test cannot be performed, snubber subcomponents that control the parameters to be verified are examined and tested. Preservice examinations are performed on snubbers after reinstallation when bench testing is used (ISTD-5224), or on snubbers where individual subcomponents are reinstalled after examination (ISTD-5225).

Defined test plan groups (DTPG) are established and the snubbers of each DTPG are tested according to an established sampling plan each fuel cycle. Sample plan size and composition is determined as required for the selected sample plan, with additional sampling as may be required for that sample plan based on test failures and failure modes identified. Snubbers that do not meet test requirements are evaluated to determine root cause of the failure, and are assigned to failure mode groups (FMG) based on the evaluation, unless the failure is considered unexplained or isolated. The number of unexplained snubber failures, not assigned to a FMG, determines the additional testing sample. Isolated failures do not require additional testing. For unacceptable snubbers, additional testing is conducted for the DTPG or FMG until the appropriate sample plan completion criteria are satisfied.

Unacceptable snubbers are adjusted, repaired, modified, or replaced. Replacement snubbers meet the requirements of ISTD-1600. Post-maintenance examination and testing, and examination and testing of repaired snubbers, is done to verify as acceptable the test parameters that may have been affected by the repair or maintenance activity.

Service life for snubbers is established, monitored and adjusted as required by ISTD-6000 and the guidance of ASME OM Code Nonmandatory Appendix F.

3.9.6 INSERVICE TESTING OF PUMPS AND VALVES

Revise the third sentence of the third paragraph of DCD Subsection 3.9.6, and add information between the third and fourth sentences as follows:

STD COL 3.9-4 The edition and addenda to be used for the inservice testing program are administratively controlled; the description of the inservice testing program in this section is based on the ASME OM Code 2001 Edition through 2003 Addenda. The initial inservice testing program incorporates 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. Limitations and modifications set forth in 10 CFR 50.55a are incorporated.

Revise the fifth sentence of the sixth paragraph of DCD Subsection 3.9.6 as follows:

STD COL 3.9-4 Alternate means of performing these tests and inspections that provide equivalent demonstration may be developed in the inservice test program ~~as described in subsection 3.9.8~~.

Revise the first two sentences of the final paragraph of DCD Subsection 3.9.6 to read as follows:

STD COL 3.9-4 A preservice test program, which identifies the required functional testing, is to be submitted to the NRC prior to performing the tests and following the start of construction. The inservice test program, which identifies requirements for functional testing, is to be submitted to the NRC prior to the anticipated date of commercial operation as described above.

Add the following text after the last paragraph of DCD Subsection 3.9.6:

Table 13.4-201 provides milestones for preservice and inservice test program implementation.

3.9.6.2.2 Valve Testing

Add the following prior to the initial paragraph of DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Valve testing uses reference values determined from the results of preservice testing or inservice testing. These tests that establish reference and IST values are performed under conditions as near as practicable to those expected during the IST. Reference values are established only when a valve is known to be operating acceptably.

Pre-conditioning of valves or their associated actuators or controls prior to IST testing undermines the purpose of IST testing and is not allowed. Pre-conditioning includes manipulation, pre-testing, maintenance, lubrication, cleaning, exercising, stroking, operating, or disturbing the valve to be tested in any way, except as may occur in an unscheduled, unplanned, and unanticipated manner during normal operation.

Add the following sentence to the end of the fourth paragraph under the heading “Manual/Power-Operated Valve Tests”:

STD COL 3.9-4 Stroke time is measured and compared to the reference value, except for valves classified as fast-acting (e.g., solenoid-operated valves with stroke time less than 2 seconds), for which a stroke time limit of 2 seconds is assigned.

Add the following paragraph after the fifth paragraph under the heading "Manual/Power-Operated Valve Tests":

STD COL 3.9-4 During valve exercise tests, the necessary valve obturator movement is verified while observing an appropriate direct indicator, such as indicating lights that signal the required changes of obturator position, or by observing other evidence or positive means, such as changes in system pressure, flow, level, or temperature that reflects change of obturator position.

Insert new second sentence of the paragraph containing the subheading “Power-Operated Valve Operability Tests” in DCD Subsection 3.9.6.2.2 (immediately following the first sentence of the DCD paragraph) to read:

STD COL 3.9-4 The POVs include the motor-operated valves.

Add the following sentence as the last sentence of the paragraph containing the subheading "Power-Operated Valve Operability Tests" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Table 13.4-201 provides milestones for the MOV program implementation.

Insert the following as the last sentence in the paragraph under the bulleted item titled "Risk Ranking" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Guidance for this process is outlined in the JOG MOV PV Study, MPR-2524-A.

Insert the following text after the last paragraph under the sub-heading of "Power-Operated Valve Operability Tests" and before the sub-heading "Check Valve Tests" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 **Active MOV Test Frequency Determination** - The ability of a valve to meet its design basis functional requirements (i.e. required capability) is verified during valve qualification testing as required by procurement specifications. Valve qualification testing measures valve actuator actual output capability. The actuator output capability is compared to the valve's required capability defined in procurement specifications, establishing functional margin; that is, that increment by which the MOV's actual output capability exceeds the capability required to operate the MOV under design basis conditions. DCD Subsection 5.4.8 discusses valve functional design and qualification requirements. The initial inservice test frequency is determined as required by ASME OM Code Case OMN-1, Revision 1 (Reference 202). The design basis capability testing of MOVs utilizes guidance from Generic Letter 96-05 and the JOG MOV Periodic Verification PV Program. Valve functional margin is evaluated following subsequent periodic testing to address potential time-related performance degradation, accounting for applicable uncertainties in the analysis. If the evaluation shows that the functional margin will be reduced to less than established acceptance criteria within the established test interval, the test interval is decreased to less than the time for the functional margin to decrease below acceptance criteria. If there is not sufficient data to determine test frequency as described above, the test frequency is limited to not exceed two (2) refueling cycles or three (3) years, whichever is longer, until sufficient data exist to extend the test frequency. Appropriate justification is provided for any increased test interval, and the maximum test interval shall not exceed 10 years. This is to ensure that each MOV in the IST program will have adequate margin (including consideration for aging-related degradation, degraded voltage, control switch repeatability, and load-sensitive MOV behavior) to remain operable until the next scheduled test, regardless of its risk categorization or safety significance. Uncertainties associated with performance of these periodic verification tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed.

Maximum torque and/or thrust (as applicable) achieved by the MOV (allowing sufficient margin for diagnostic equipment inaccuracies and control switch repeatability) are established so as not to exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV.

Solenoid-operated valves (SOVs) are tested to confirm the valve moves to its energized position and is maintained in that position, and to confirm that the valve moves to the appropriate failure mode position when de-energized.

Other Power-Operated Valve Operability Tests - Power-Operated valves other than active MOVs are exercised quarterly in accordance with ASME OM ISTC, unless justification is provided in the inservice testing program for testing these valves at other than Code mandated frequencies.

Although the design basis capability of power-operated valves is verified as part of the design and qualification process, power-operated valves that perform an active safety function are tested again after installation in the plant, as required, to ensure valve setup is acceptable to perform their required functions, consistent with valve qualification. These tests, which are typically performed under static (no flow or pressure) conditions, also document the "baseline" performance of the valves to support maintenance and trending programs. During the testing, critical parameters needed to ensure proper valve setup are measured. Depending on the valve and actuator type, these parameters may include seat load, running torque or thrust, valve travel, actuator spring rate, bench set and regulator supply pressure. Uncertainties associated with performance of these tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed.

Additional testing is performed as part of the air-operated valve (AOV) program, which includes the key elements for an AOV Program as identified in the JOG AOV program document, Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000 (References 203 and 204). The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions, by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of air- and other power-operated valves included in the IST program. For example, key lessons learned addressed in the AOV program include:

- Valves are categorized according to their safety significance and risk ranking.
- Setpoints for AOVs are defined based on current vendor information or valve qualification diagnostic testing, such that the valve is capable of performing its design-basis function(s).
- Periodic static testing is performed, at a minimum on high risk (high safety significance) valves, to identify potential degradation, unless those valves are periodically cycled during normal plant operation, under conditions that meet or exceed the worst case operating conditions within the licensing basis of the plant for the valve, which would provide adequate periodic demonstration of AOV capability. If required based on valve qualification or

operating experience, periodic dynamic testing is performed to re-verify the capability of the valve to perform its required functions.

- Sufficient diagnostics are used to collect relevant data (e.g., valve stem thrust and torque, fluid pressure and temperature, stroke time, operating and/or control air pressure, etc.) to verify the valve meets the functional requirements of the qualification specification.

- Test frequency is specified, and is evaluated each refueling outage based on data trends as a result of testing. Frequency for periodic testing is in accordance with References 203 and 204, with a minimum of 5 years (or 3 refueling cycles) of data collected and evaluated before extending test intervals.

- Post-maintenance procedures include appropriate instructions and criteria to ensure baseline testing is re-performed as necessary when maintenance on the valve, repair or replacement, have the potential to affect valve functional performance.

- Guidance is included to address lessons learned from other valve programs specific to the AOV program.

- Documentation from AOV testing, including maintenance records and records from the corrective action program are retained and periodically evaluated as a part of the AOV program.

Insert the following paragraph as the last paragraph under the sub-heading of “Power-Operated Valve Operability Tests” (following the previously added paragraph) and just before the sub-heading “Check Valve Tests” in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Successful completion of the preservice and IST of MOVs, in addition to MOV testing as required by 10 CFR 50.55a, demonstrates 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 a 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.

Add the paragraph below as the last paragraph of FSAR Subsection 3.9.6.2.2 prior to the subheading “Check Valve Tests”:

STD COL 3.9-4 The attributes of the AOV testing program described above, to the extent that they apply to and can be implemented on other safety-related power-operated valves, such as electro-hydraulic valves, are applied to those other power-operated valves.

Add the following new paragraph under the heading "Check Valves Tests" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Preoperational testing is performed during the initial test program (refer to DCD Subsection 14.2) to verify that valves are installed in a configuration that allows correct operation, testing, and maintenance. Preoperational testing verifies that piping design features accommodate check valve testing requirements. Tests also verify disk movement to and from the seat and determine, without disassembly, that the valve disk positions correctly, fully opens or fully closes as expected, and remains stable in the open position under the full spectrum of system design-basis fluid flow conditions.

Add the following new last paragraphs under the subheading "Check Valve Exercise Tests" in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Acceptance criteria for this testing consider the specific system design and valve application. For example, a valve's safety function may require obturator movement in both open and closed directions. A mechanical exerciser may be used to operate a check valve for testing. Where a mechanical exerciser is used, acceptance criteria are provided for the force or torque required to move the check valve's obturator. Exercise tests also detect missing, sticking, or binding obturators.

When operating conditions, valve design, valve location, or other considerations prevent direct observation or measurements by use of conventional methods to determine adequate check valve function, diagnostic equipment and nonintrusive techniques are used to monitor internal conditions. Nonintrusive tests used are dependent on system and valve configuration, valve design and materials, and include methods such as ultrasonic (acoustic), magnetic, radiography, and use of accelerometers to measure system and valve operating parameters (e.g., fluid flow, disk position, disk movement, disk impact, and the presence or absence of cavitation and back-tapping). Nonintrusive techniques also detect valve degradation. Diagnostic equipment and techniques used for valve operability determinations are verified as effective and accurate under the PST program.

Testing is performed, to the extent practicable, under normal operation, cold shutdown, or refueling conditions applicable to each check valve. Testing includes effects created by sudden starting and stopping of pumps, if applicable, or other conditions, such as flow reversal. When maintenance that could affect valve performance is performed on a valve in the IST program, post-maintenance testing is conducted prior to returning the valve to service.

Add the following new paragraph under the heading "Other Valve Inservice Tests" following the Explosively Actuated Valves paragraph in DCD Subsection 3.9.6.2.2:

STD COL 3.9-4 Industry and regulatory guidance is considered in development of IST program for squib valves. In addition, the IST program for squib valves incorporates 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.

3.9.6.2.3 Valve Disassembly and Inspection

Add the following paragraph as the new second paragraph of DCD Subsection 3.9.6.2.3:

STD COL 3.9-4 During the disassembly process, the full-stroke motion of the obturator is verified. Nondestructive examination is performed on the hinge pin to assess wear, and seat contact surfaces are examined to verify adequate contact. Full-stroke motion of the obturator is re-verified immediately prior to completing reassembly. At least one valve from each group is disassembled and examined at each refueling outage, and all the valves in each group are disassembled and examined at least once every eight years. Before being returned to service, valves disassembled for examination or valves that received maintenance that could affect their performance are exercised with a full- or part-stroke. Details and bases of the sampling program are documented and recorded in the test plan.

Add Subsections 3.9.6.2.4 and 3.9.6.2.5 following the last paragraph of DCD Subsection 3.9.6.2.3:

3.9.6.2.4 Valve Preservice Tests

STD COL 3.9-4 Each valve subject to inservice testing is also tested during the preservice test period. Preservice tests are conducted under conditions as near as practicable to those expected during subsequent inservice testing. Valves (or the control system) that have undergone maintenance that could affect performance, and valves that have been repaired or replaced, are re-tested to verify performance parameters that could have been affected are within acceptable limits. Safety and relief valves and nonreclosing pressure relief devices are preservice tested in accordance with the requirements of the ASME OM Code, Mandatory Appendix I.

Preservice tests for valves are performed in accordance with ASME OM, ISTC-3100.

3.9.6.2.5 Valve Replacement, Repair, and Maintenance

Testing in accordance with ASME OM, ISTC-3310 is performed after a valve is replaced, repaired, or undergoes maintenance. When a valve or its control system has been replaced, repaired, or has undergone maintenance that could affect valve performance, a new reference value is determined, or the previous value is reconfirmed by an inservice test. This test is performed before the valve is returned to service, or immediately if the valve is not removed from service. Deviations between the previous and new reference values are identified and analyzed. Verification that the new values represent acceptable operation is documented.

3.9.6.3 Relief Requests

Insert the following text after the first paragraph in DCD Subsection 3.9.6.3:

STD COL 3.9-4

The IST Program described herein utilizes Code Case OMN-1, Revision 1, "Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants" (Reference 202). Code Case OMN-1 establishes alternate rules and requirements for preservice and inservice testing to assess the operational readiness of certain motor operated valves in lieu of the requirements set forth in ASME OM Code Subsection ISTC.

OMN-1, Alternative Rules for the Preservice and Inservice Testing of Certain MOVs

Code Case OMN-1, Revision 1, "Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in Light Water Reactor Power Plants," establishes alternate rules and requirements for preservice and inservice testing to assess the operational readiness of certain motor-operated valves in lieu of the requirements set forth in OM Code Subsection ISTC. However, Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," June 2003, has not yet endorsed OMN-1, Revision 1.

Code Case OMN-1, Revision 0, has been determined by the NRC to provide an acceptable level of quality and safety when implemented in conjunction with the conditions imposed in Regulatory Guide 1.192. NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants," recommends the implementation of OMN-l by all licensees. Revision 1 to OMN-1 represents an improvement over Revision 0, as published in the ASME OM-2004 Code. OMN-1 Revision 1 incorporates the guidance on risk-informed testing of MOVs from OMN-11, "Risk-Informed Testing of Motor-Operated Valves," and provides additional guidance on design basis verification testing and functional margin, which eliminates the need for the figures on functional margin and test intervals in Code Case OMN-1.

The IST Program implements Code Case OMN-1, Revision 1, in lieu of the stroke-time provisions specified in ISTC-5120 for MOVs, consistent with the guidelines provided in NUREG-1482, Revision 1, Section 4.2.5.

Regulatory Guide 1.192 states that licensees may use Code Case OMN-1, Revision 0, in lieu of the provisions for stroke-time testing in Subsection ISTC of the 1995 Edition up to and including the 2000 Addenda of the ASME OM Code when applied in conjunction with the provisions for leakage rate testing in ISTC-3600 (1998 Edition with the 1999 and 2000 Addenda). Licensees who choose to apply OMN-1 are required to apply all of its provisions. The IST program incorporates the following provisions from Regulatory Guide 1.192:

(1) The adequacy of the diagnostic test interval for each motor-operated valve (MOV) is evaluated and adjusted as necessary, but not later than 5 years or three refueling outages (whichever is longer) from initial implementation of OMN-1.

(2) The potential increase in CDF and risk associated with extending high risk MOV test intervals beyond quarterly is determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

(3) Risk insights are 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.

(4) Consistent with the provisions specified for Code Case OMN-11 the potential increase in CDF and risk associated with extending high risk MOV test intervals beyond quarterly is determined to be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

Compliance with the above items is addressed in Section 3.9.6.2.2. Code Case OMN-1, Revision 1, is considered acceptable for use with OM Code-2001 Edition with 2003 Addenda. Finally, consistent with Regulatory Guide 1.192, the benefits of performing any particular test are balanced against the potential adverse effects placed on the valves or systems caused by this testing.

3.9.8 COMBINED LICENSE INFORMATION

3.9.8.2 Design Specifications and Reports

Add the following text after the second paragraph in DCD Subsection 3.9.8.2.

STD COL 3.9-2 Design specifications and design reports for ASME Section III piping are made available for NRC review. Reconciliation of the as-built piping (verification of the thermal cycling and stratification loading 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 (in accordance with DCD Tier 1 Section 2 ITAAC line item for the applicable systems).

3.9.8.3 Snubber Operability Testing

STD COL 3.9-3 This COL Item is addressed in Subsection 3.9.3.4.4.

3.9.8.4 Valve Inservice Testing

STD COL 3.9-4 This COL Item is addressed in Subsections 3.9.6, 3.9.6.2.2, 3.9.6.2.4, 3.9.6.2.5, and 3.9.6.3.

3.9.8.5 Surge Line Thermal Monitoring

STD COL 3.9-5 This COL item is addressed in Subsection 3.9.3.1.2 and Subsection 14.2.9.2.22.

3.9.8.7 As-Designed Piping Analysis

Add the following text at the end of DCD Subsection 3.9.8.7.

STD COL 3.9-7 The as-designed piping analysis is provided for the piping lines chosen to demonstrate all aspects of the piping design. A design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class 1 piping using the methods and criteria outlined in DCD Table 3.9-19 is made available for NRC review.

This COL item is also addressed in Subsection 14.3.3.

3.9.9 REFERENCES

201. Not used.

202. ASME Code Case OMN-1, Revision 1, "Alternative Rules for the Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants."

203. Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000.

204. USNRC, Eugene V. Imbro, letter to Mr. David J. Modeen, Nuclear Energy Institute, Comments on Joint Owners' Group Air Operated Valve Program Document, dated October 8, 1999.

STD SUP 3.9-3

Table 3.9-201
Safety Related Snubbers

System	Snubber (Hanger) No.	Line #
CVS	APP-CVS-PH-11Y0164	L001
PXS	APP-PXS-PH-11Y0020	L021A
RCS	APP-RCS-PH-11Y0039	L215
RCS	APP-RCS-PH-11Y0067	L005B
RCS	APP-RCS-PH-11Y0080	L112
RCS	APP-RCS-PH-11Y0081	L215
RCS	APP-RCS-PH-11Y0082	L112
RCS	APP-RCS-PH-11Y0090	L118A
RCS	APP-RCS-PH-11Y0099	L022B
RCS	APP-RCS-PH-11Y0103	L003
RCS	APP-RCS-PH-11Y0105	L003
RCS	APP-RCS-PH-11Y0112	L032A
RCS	APP-RCS-PH-11Y0429	L225B
RCS	APP-RCS-PH-11Y0528	L005A
RCS	APP-RCS-PH-11Y0539	L225C
RCS	APP-RCS-PH-11Y0550	L011B
RCS	APP-RCS-PH-11Y0551	L011A
RCS	APP-RCS-PH-11Y0553	L153B
RCS	APP-RCS-PH-11Y0555	L153A
RCS	APP-RCS-PH-11Y2005	L022A
RCS	APP-RCS-PH-11Y2101	L032B
RCS	APP-RCS-PH-11Y2117	L225A

System	Snubber (Hanger) No.	Line #
RNS	APP-RNS-PH-12Y2060	L006
SGS	APP-SGS-PH-11Y0001	L003B
SGS	APP-SGS-PH-11Y0002	L003B
SGS	APP-SGS-PH-11Y0004	L003B
SGS	APP-SGS-PH-11Y0057	L003A
SGS	APP-SGS-PH-11Y0058	L004B
SGS	APP-SGS-PH-11Y0063	L003A
SGS	APP-SGS-PH-11Y0065	L005B
SGS	APP-SGS-PH-12Y0136	L015C
SGS	APP-SGS-PH-12Y0137	L015C
SGS	APP-SGS-PH-11Y0470	L006B
SGS	APP-SGS-PH-11Y2002	L006A
SGS	APP-SGS-PH-11Y2021	L006A
SGS	APP-SGS-PH-11Y3101	L006B
SGS	APP-SGS-PH-11Y3102	L006B
SGS	APP-SGS-PH-11Y3121	L006B
SGS	APP-SGS-PH-11Y0463	L006A
SGS	APP-SGS-PH-11Y0464	L006A
SGS	SG 1 Snubber A (1A)	(1)
SGS	SG 1 Snubber B (1B)	(1)
SGS	SG 2 Snubber A (2A)	(1)
SGS	SG 2 Snubber B (2B)	(1)

(1) These snubbers are on the upper lateral support assembly of the steam generators.

3.10 SEISMIC AND DYNAMIC QUALIFICATION OF SEISMIC CATEGORY I MECHANICAL AND ELECTRICAL EQUIPMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

3.11 ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

3.11.5 COMBINED LICENSE INFORMATION ITEM FOR EQUIPMENT QUALIFICATION FILE

Add the following text to the end of DCD Subsection 3.11.5.

STD COL 3.11-1 The COL holder is responsible for the maintenance of the equipment qualification file upon receipt from the reactor vendor. The documentation necessary to support the continued qualification of the equipment installed in the plant that is within the Environmental Qualification (EQ) Program scope is available in accordance with 10 CFR Part 50 Appendix A, General Design Criterion 1.

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 contents of the qualification files are discussed in DCD Section 3D.7. The files are maintained for the operational life of the plant.

For equipment not located in a harsh environment, design specifications received from the reactor vendor are retained. Any plant modifications that impact the equipment use the original specifications for modification or procurement. This process is governed by applicable plant design control or configuration control procedures.

Central to the EQ Program is the EQ Master Equipment List (EQMEL). This EQMEL identifies the electrical and mechanical equipment or components that must be environmentally qualified for use in a harsh environment. The EQMEL consists of equipment that is essential to emergency reactor shutdown, containment isolation, reactor core cooling, or containment and reactor heat removal, or that is otherwise essential in preventing significant release of radioactive material to the environment. This list is developed from the equipment list provided in AP1000 DCD Table 3.11-1. 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. To delete a component from the EQ Program, a deletion justification is prepared that demonstrates why the component can be deleted.

This justification consists of an analysis of the component, an associated circuit review if appropriate, and a safety evaluation. The justification is released and/or referenced on an appropriate change document. For changes to the EQMEL, supporting documentation is completed and approved prior to issuing the changes. This documentation includes safety reviews and new or revised EQ files. 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. These reviews address EQ issues associated with the activity. 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.

Engineering change documents or maintenance documents generated to document work performed on an EQ component, which may not have an impact on the EQ file, are reviewed against the current revision of the EQ files for potential impact. Changes to EQ documentation may be due to, but not limited to, plant modifications, calculations, corrective maintenance, or other EQ concerns.

Table 13.4-201 provides milestones for EQ implementation.

APPENDIX 3A
HVAC DUCTS AND DUCT SUPPORTS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3B
LEAK-BEFORE-BREAK EVALUATION OF THE AP1000 PIPING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3C
REACTOR COOLANT LOOP ANALYSIS METHODS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3D
METHODOLOGY FOR QUALIFYING AP1000 SAFETY-RELATED ELECTRICAL AND MECHANICAL EQUIPMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3E
HIGH-ENERGY PIPING IN THE NUCLEAR ISLAND

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3F
CABLE TRAYS AND CABLE TRAY SUPPORTS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3G
NUCLEAR ISLAND SEISMIC ANALYSES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

VEGP SUP 3.7-3 Appendix 3GG is provided to supplement the information in DCD Appendix 3G.

APPENDIX 3GG

VEGP SUP 3.7-3 3-D SSI ANALYSIS OF AP1000 AT VOGTLE SITE USING NI15 MODEL

3-D SSI ANALYSIS OF AP1000 AT VOGTLE SITE USING NI15 MODEL

FOR

VEGP UNITS 3 AND 4

October 2010

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1. Introduction

This report presents the results of the three-dimensional soil-structure interaction (SSI) analysis of the AP1000 plant at the Vogtle site to confirm the applicability of the AP1000 design to the site.

This report supplements the two-dimensional site-specific SSI analysis previously submitted as Appendix 2.5E in the Vogtle Early Site Permit Application.

Site-specific SSI analysis is required since the site specific design response spectra exceed the certified seismic design response spectra (CSDRS) at some limited frequency range and the Vogtle soil profile is significantly different than the AP1000 generic soil profiles in shear wave velocity versus depth and overall soil depth.

Reference 1 describes changes to the AP1000 NI20 SASSI model now identified as NI20r and provides revised AP1000 CSDRS broadened envelope ISRS. This report reflects those changes and consists of updating the Vogtle NI15 SASSI model, rerunning the Vogtle SASSI analyses using the updated Vogtle NI15 SASSI model to generate revised Vogtle ISRS at the six key locations for the Vogtle soil profile (Lower Bound, Best Estimate, and Upper Bound soil cases), and providing a comparison of the revised Vogtle ISRS to the new AP1000 CSDRS broadened envelope ISRS.

2. Methodology

The free-field analyses are performed using the Bechtel Computer Program SHAKE2000. The SSI analyses are performed using the Bechtel Computer Program SASSI2000

3. Vogtle Site Profile

A detailed description of the site geology and soil stratigraphy including the extent and characteristics of the backfill materials is contained in the Early Site Permit Application and is not repeated in this report. For the three-dimensional SSI analysis, the same soil profiles used for the two-dimensional SSI analysis are used. The strain-compatible soil shear-wave velocity and damping profiles for the three soil cases, (upper bound (UB), median (BE) and lower bound (LB)) are shown in Figure 1 and Figure 2. Note that the UB shear-wave velocity profile is combined with the LB damping profile to form the UB SSI soil profile. Likewise, the LB velocity profile is combined with the UB damping profile to form the LB SSI soil profile. The BE shear wave velocity and damping profiles are for the BE SSI soil profile. These profiles are obtained from the group of simulated soil profiles used for development of the soil amplification factors and site specific ground motions by considering the median and one standard deviation of the range of data and incorporating the NUREG-0800 requirement of the minimum soil shear modulus variation of 1.5. For SSI analysis, the rock was modeled at the depth of about 1000 ft corresponding to the approximate depth of the rock at the site.

For comparison purposes, the strain-compatible generic soil profiles used for certified design of AP1000 are compared with the strain-compatible Vogtle UB, BE and LB site-specific soil profiles in Figure 3. As shown, the Vogtle site-specific soil profiles are softer than the lower-bound generic Soft Soil profile in the upper 50 ft. In addition, the Vogtle site-specific soil profiles extend to the depth of about 1000 ft whereas the generic soil profiles are only 120 ft deep overlying a bedrock layer assumed to be a halfspace layer below 120 ft depth.

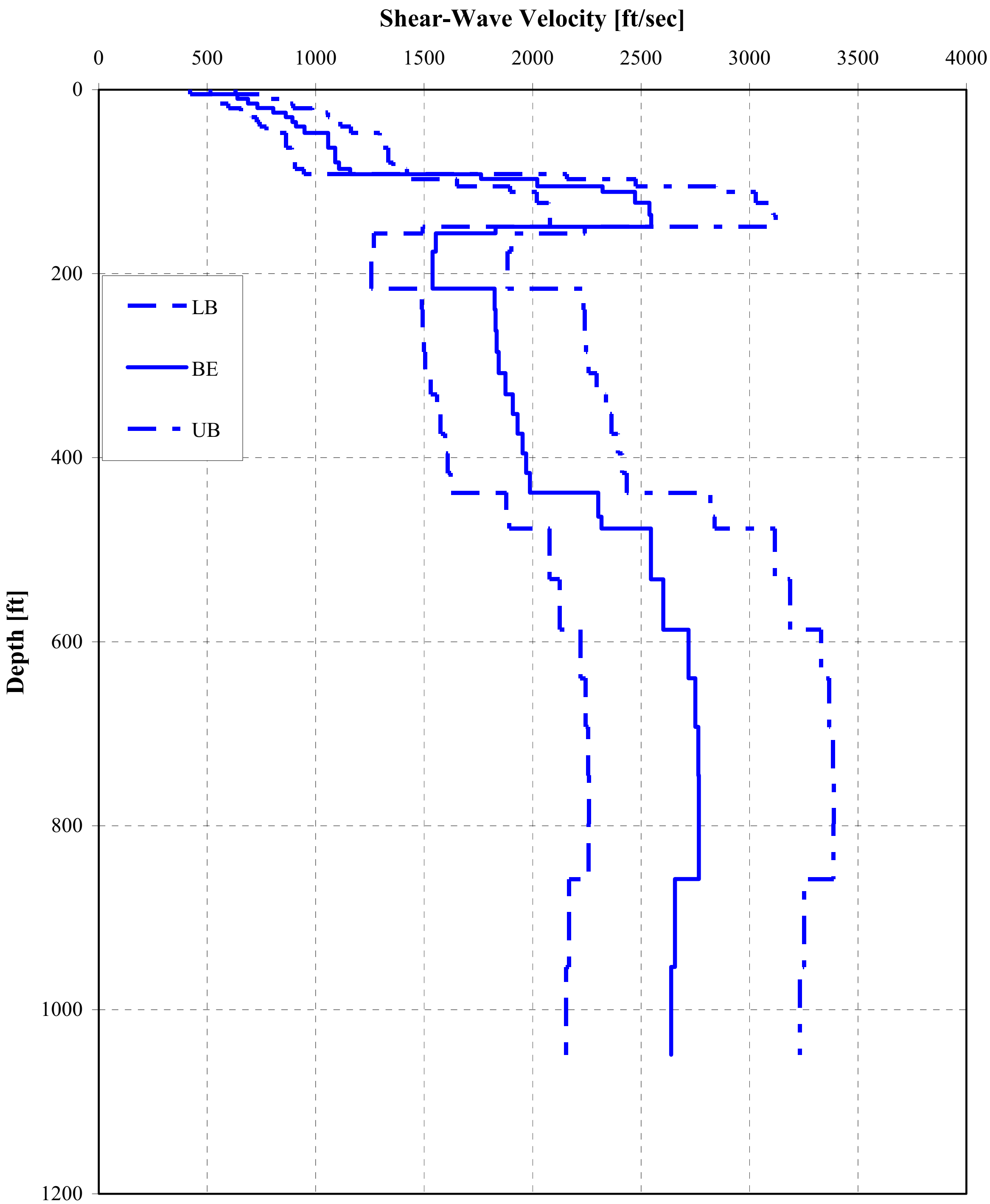

Figure 1 Vogtle Strain-Compatible Soil Shear Wave Velocity Profiles Used in SSI Analysis

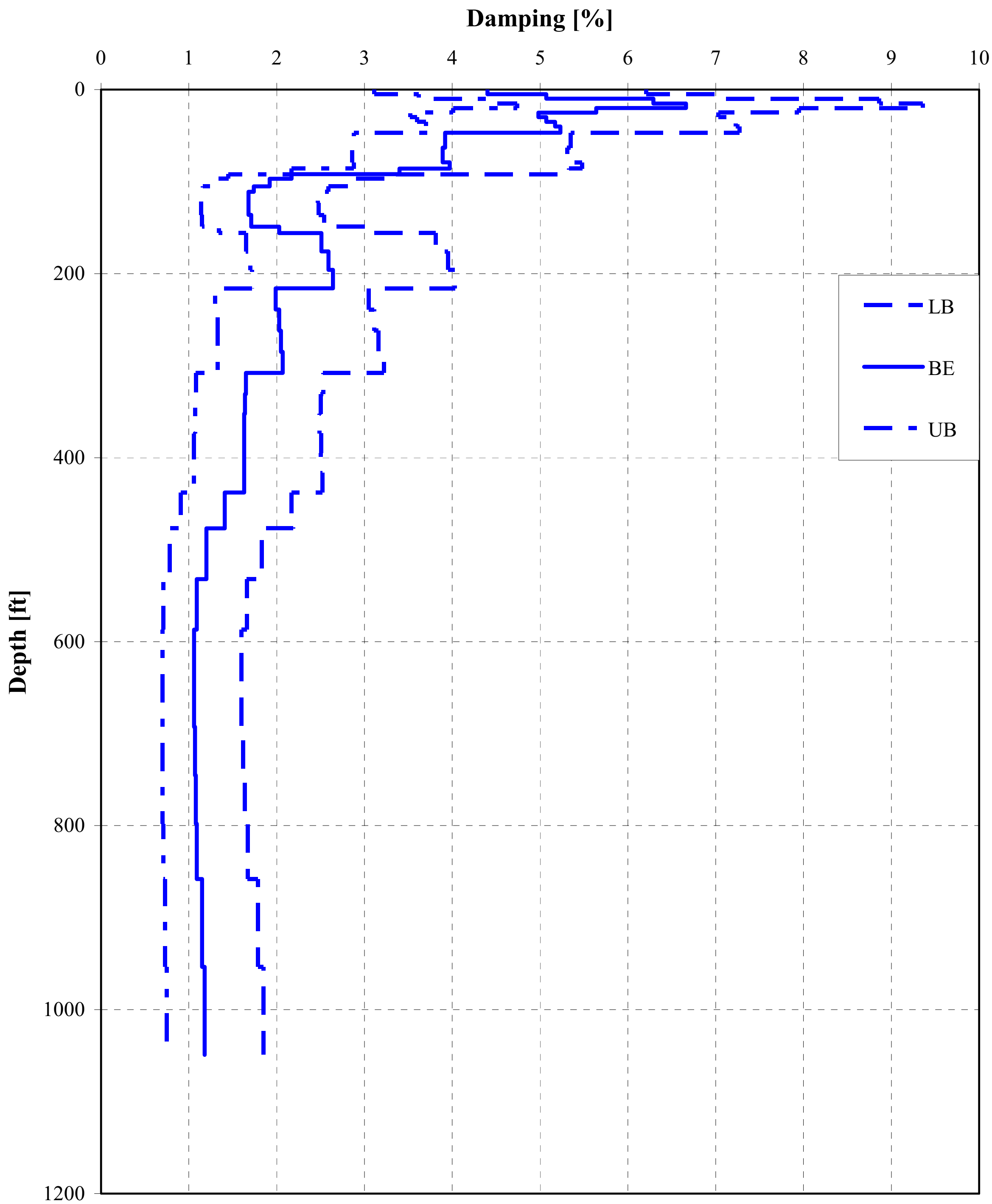

Figure 2 Vogtle Strain-Compatible Soil Damping Profiles Used in SSI Analysis

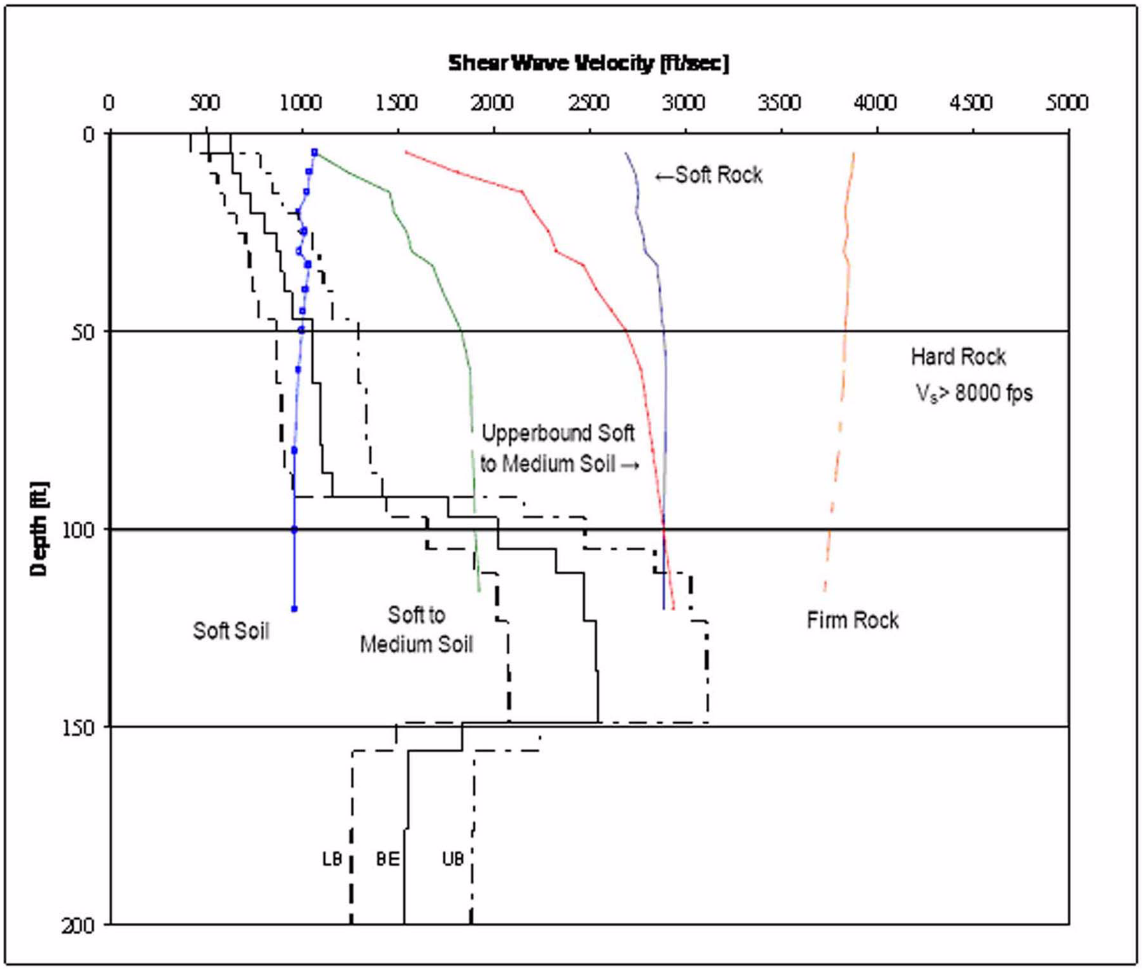

Figure 3 Vogtle Site-Specific and AP1000 Generic Strain-Compatible Soil Profiles

4. Vogtle Site Specific Seismic Motion

As described in the ESP application, the ground motion response spectra (GMRS) at the Vogtle site are defined at the finished grade at the top of the backfill. The foundation input response spectra (FIRS) is at the foundation horizon at the depth of 40 ft below the finished grade. FIRS and GMRS are compared with CSDRS in Figure 4 and Figure 5 for the horizontal and vertical motions, respectively. Note that the FIRS is an outcrop motion at the foundation level obtained from the soil column analysis of the site full soil column extending to the top of the backfill.

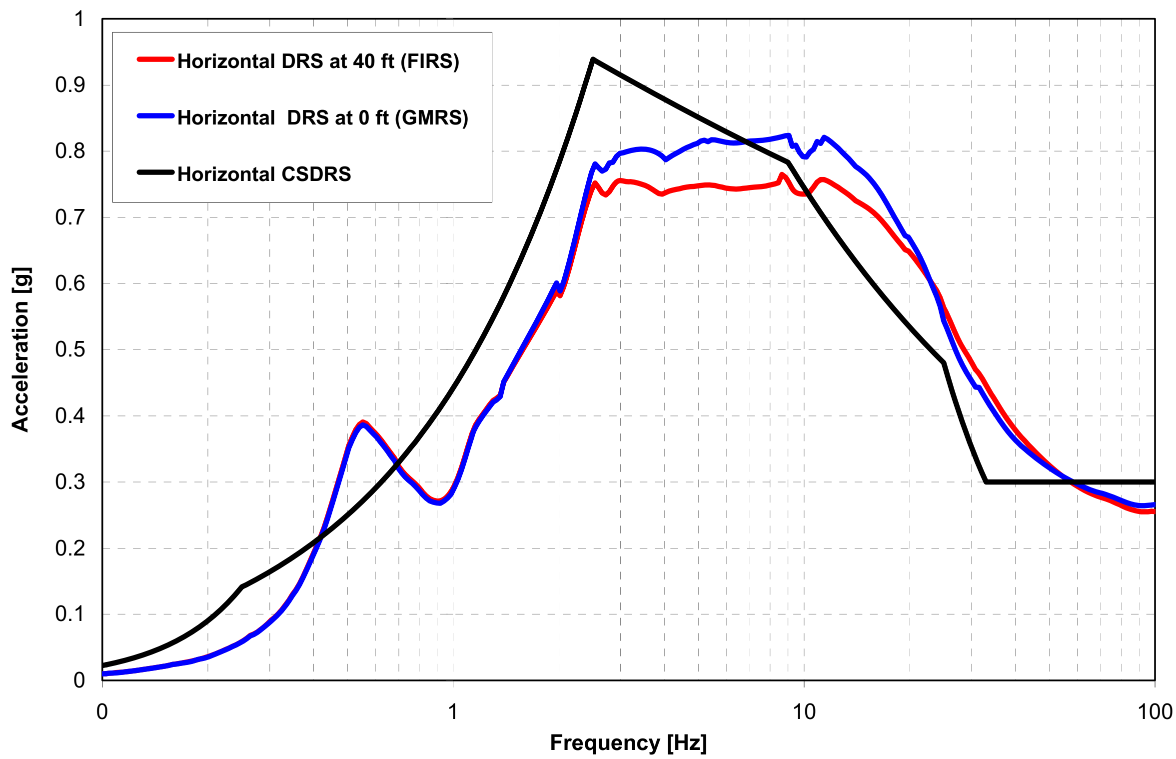

Figure 4 AP1000 CSDRS and Vogtle GMRS and FIRS - Horizontal Motion (5% Damping)

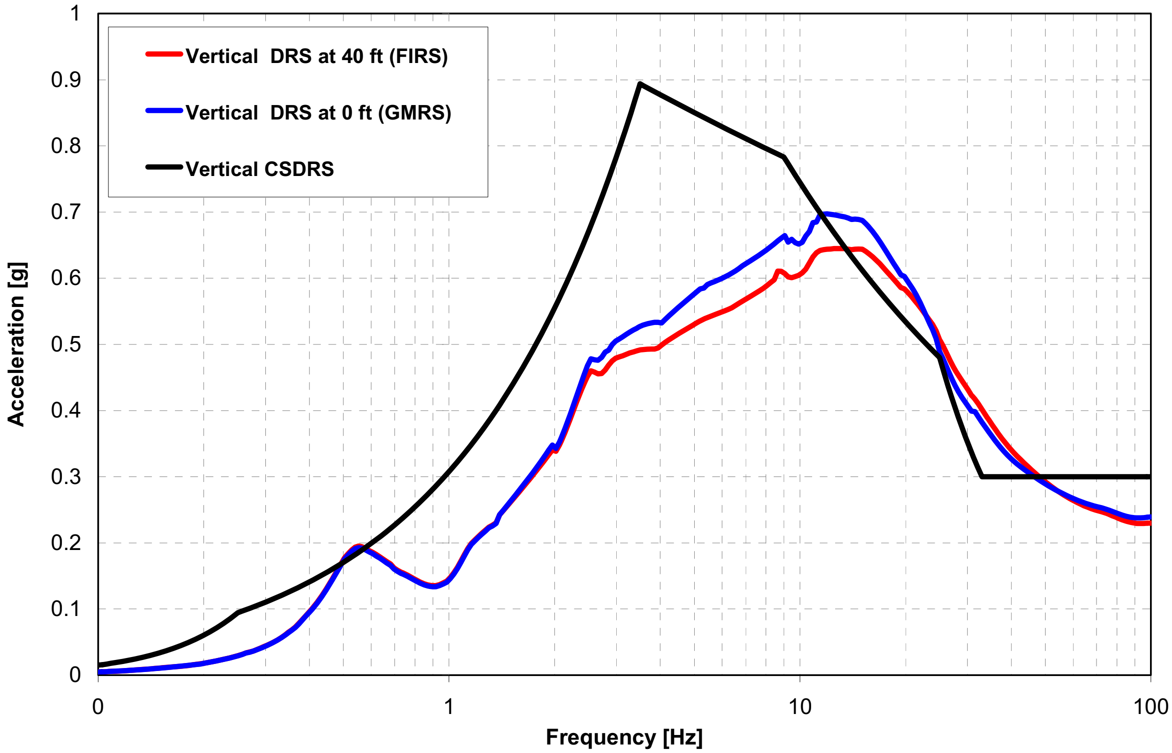

Figure 5 AP1000 CSDRS and Vogtle GMRS and FIRS - Vertical Motion (5% Damping)

As shown in the above figures, both the horizontal and vertical Design Response Spectra (DRS) at both GMRS and FIRS levels exceed the CSDRS at a limited frequency range.

4.1 SSI Input Motion

The development of SSI input motion follows the procedure outlined in the recent NRC position on this subject (ADAMS Accession Numbers ML083580072 and ML083020171). The development of SSI input motion is consistent with the development of FIRS and the required check has been made at the ground surface to evaluate the adequacy of the SSI input motion. Using the three SSI soil profiles defined above, acceleration time histories compatible with the FIRS are generated and applied as outcrop input motion at the depth of 40 ft, and the response motions at the surface are computed using Bechtel Program SHAKE2000. The resulting three spectra are compared with the surface design spectra (GMRS) in Figure 6 through Figure 8 for the horizontal H1, H2 and the Vertical component of the motion, respectively. As shown in these figures, the envelope of the three horizontal SSI input motions (LB, BE and UB) adequately envelops the GMRS in the two horizontal directions (H1 and H2) and no further modification of the horizontal motion is warranted, The vertical motions, however, are slightly less that the vertical GMRS in the frequency range 2.5 to 7 Hz. For this reason the vertical time history associated with the lower bound soil profile analysis was increased uniformly by a factor of 1.11. Figure 9 shows the comparison for the vertical motion confirming the enveloping spectra from the three soil profiles envelop the vertical GMRS at the ground surface.

For SASSI SSI analysis and for each SSI soil profile, the outcrop motions were converted to in-column motions at the depth of 40 ft and the in-column motions are subsequently used in the SSI analysis. For each of the three soil profiles, three in-column time histories are developed resulting in a total of nine incolumn time histories for SSI analysis. As described above, the vertical in-column time history corresponding to the LB soil profile was increased by a factor of 1.11 to meet the enveloping requirement at the surface.

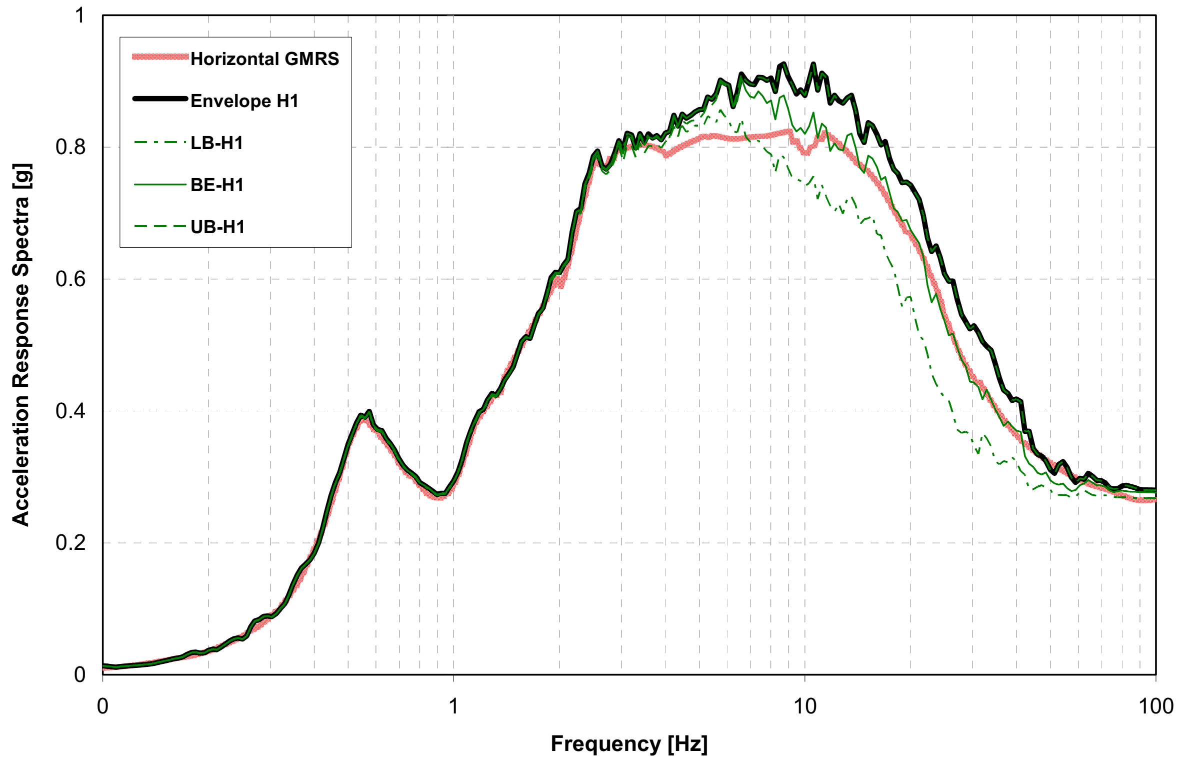

Figure 6 Comparison of H1 Response Motion and GMRS at the Ground Surface Level (5% Damping)

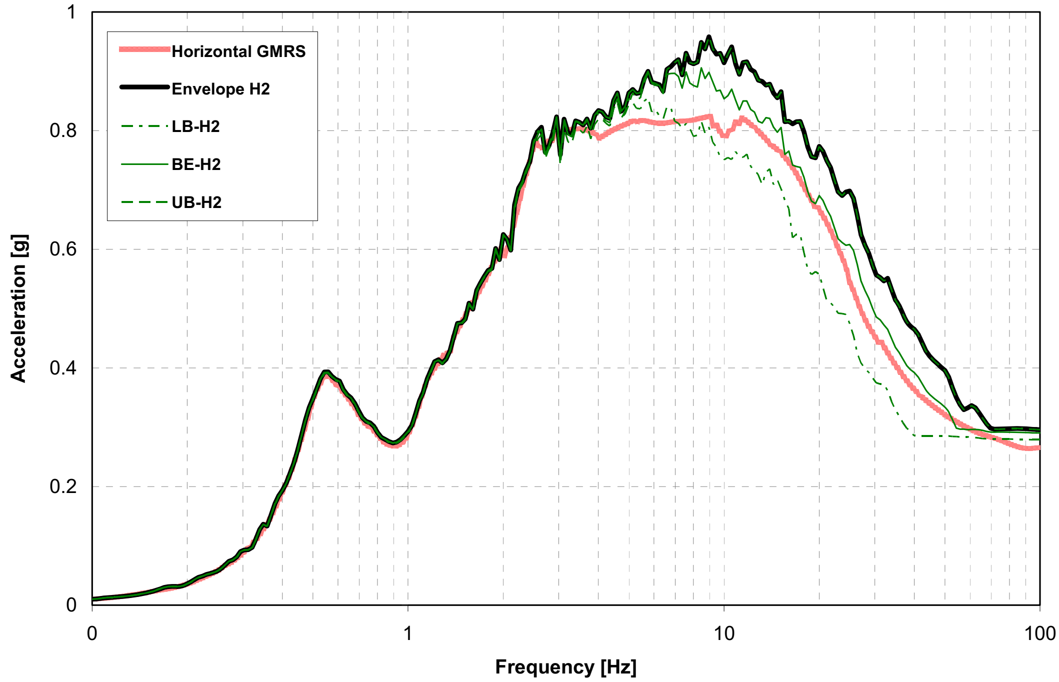

Figure 7 Comparisons of the H2 Response Motions with the GMRS at the Ground Surface Level (5% Damping)

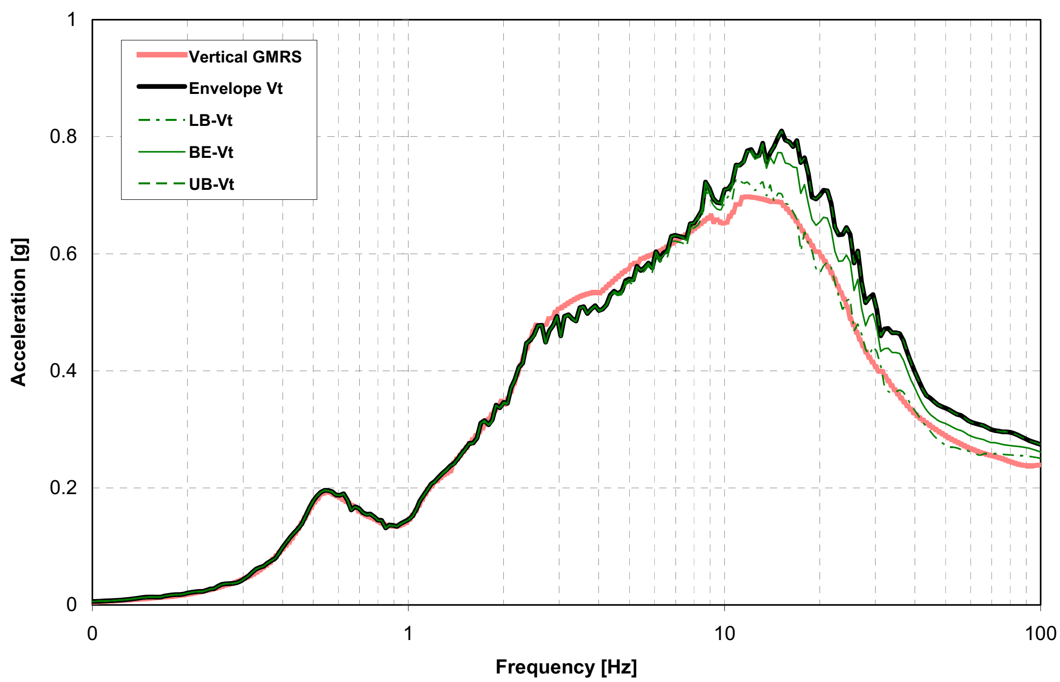

Figure 8 Comparisons of the Vertical Response Motions with the GMRS at the Ground Surface Level (5% Damping)

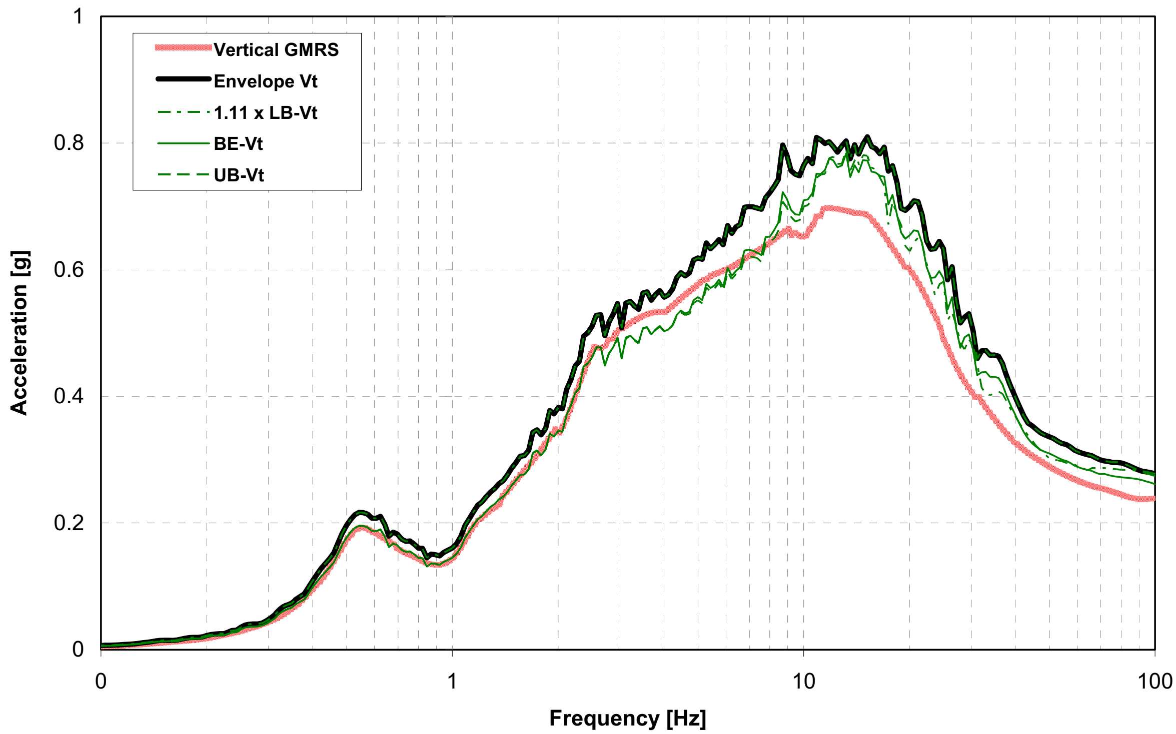

Figure 9 Comparisons of the Modified Vertical Response Motions with the GMRS at the Ground Surface Level (5% Damping)

5. Structural Model

The AP1000 model used for Vogtle site-specific SSI analysis is a three-dimensional finite element model defined as the NI15 model that is developed by Westinghouse. This model was developed specifically for the Vogtle site to incorporate additional refinement in order to capture the Vogtle high frequency exceedance beyond CSDRS as shown in Figure 4 and Figure 5. In addition as shown in Figure 3, the Vogtle soil profile is softer than the generic profiles in the upper 50 ft and significantly deeper with an inverted impedance mismatch below the Blue Bluff marl requiring site specific modeling and analysis to evaluate applicability of the design.

The AP1000 Nuclear Island consists of the Auxiliary and Shield building (ASB), Containment Internal Structure (CIS), Reactor Coolant Loop and Steel Containment Vessel (SCV). The ANSYS NI15 Model, averaging 15' by 15' for solid and shell elements in the ASB, is shown in Figure 10. The structure model has over 6300 nodes and 7500 elements. The embedded part of the NI is modeled with 5 layers of elements for a total embedment depth of 39.5 ft. Solid elements and Beam elements for SCV, CIS including Reactor Coolant Loop, Pressurizer, and polar crane are shown in Figure 11.

The NI15 was verified by Westinghouse by assuring that the mass distribution, the modal behavior and the floor response spectra results were consistent in ANSYS with WEC's most detailed model which is the model used for Hard Rock (NI10). The mass, centroid, and moment of inertia analysis determined the geometric and material properties were consistent with the finite element model NI10. The dynamic behavior of the Nuclear Island building is identified by means of a modal analysis, and a floor response spectra comparison of the two models.

The ANSYS NI15 model is converted into the SASSI NI15 Model where excavated soil elements are added. The SASSI NI15 model is used in the Soil and Structure Interaction (SSI) analysis.

Due to the changes to the AP1000 NI20 SASSI model now identified as NI20r as described in Reference 1, the Vogtle NI15 SASSI model was revised from that described above as follows:

1. The properties of the Shield Building walls and air-inlet were updated to reflect the Shield Building design changes.

2. Modeling corrections to the Westinghouse AP1000 NI20 SASSI, as described in Reference 1, Section 4.2.3 “Corrections to NI20 SASSI Model”, were not required for the Vogtle NI15 SASSI model. These corrections to the SASSI NI20 model were to address modeling concerns with beam to solid element connectivity and improve the stress distribution in the basemat. The Vogtle NI15 SASSI model beam to solid element connectivity already properly modeled the connections between the solid elements and beam elements. Unlike the NI20 SASSI model that modeled the Auxiliary Building portion of the basemat of the Nuclear Island as shell elements, the Vogtle NI15 SASSI model used solid elements for the entire basemat. Therefore, there was no issue with the stress distribution at the basemat interface between the Auxiliary Building and the Containment Internal Structure (CIS).

3. The original NI20 SASSI model was revised to account for stiffness due to out-of-plane flexure where the walls, which are modeled as shell elements, connect to the floors, which are modeled as solid elements. Therefore, the Vogtle NI15 SASSI model was revised by extending the wall shell elements the depth of one solid element to capture the effects of out-of-plane flexural stiffness. This modeling change showed no significant effect on the response since in-plane wall stiffness was the controlling contributor to overall lateral structural stiffness.

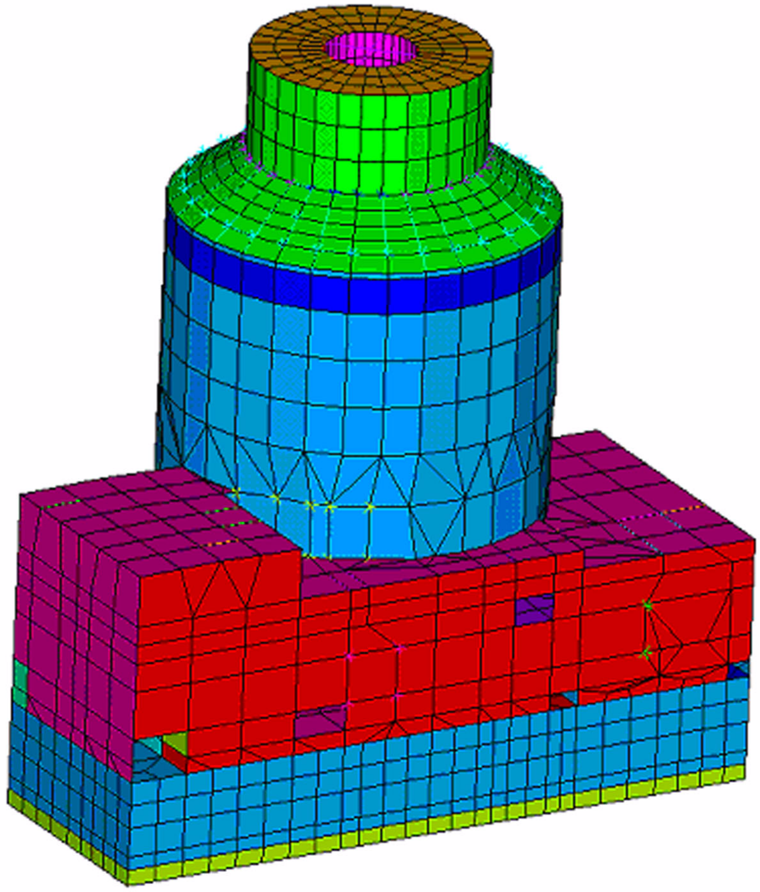

Figure 10 NI15 Finite Element Model

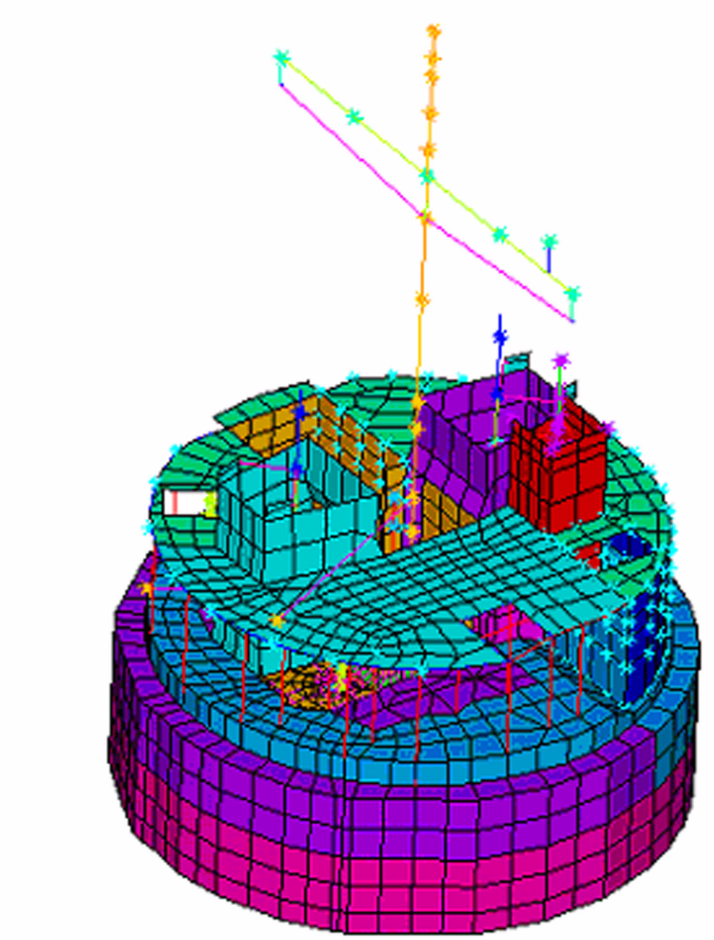

Figure 11 NI15 CIS, RCL, and SCV Elements

6. SSI Analysis and Results

Using the above structural model, the Vogtle site-specific SASSI SSI model of AP1000 was constructed by modeling the soil profile and the soil foundation model for the embedded part of the nuclear island (NI). For all structural members, 4% material damping was used. This damping is considered to be conservative and is representative of the lower bound value for damping compatible to structural response per RG 1.61. For each soil profile, the respective in-column motions were used as input at the depth of the foundation level with excitation in all three (North-South, East-West, and Vertical) directions. The results in terms of in-structure response spectra (ISRS) at 5% damping at the six key locations in the NI (Table 1) are computed. The coupling responses are combined using the SRSS method. The analyses are performed to 30 Hz (15 Hz for LB, 17 Hz for BE, 30 Hz for UB) to cover all frequencies of interest for the given design motion.

Table 1: Key Location for ISRS Comparison with DCD

Node	X* [ft]	Y* [ft]	Z [ft]	Location
10115	1116.5	948.5	116.5	ASB NE Corner at Control Room Floor
11111	929	1000	179.19	ASB Corner of Fuel Building Roof at Shield Building
12052	956.5	1000	327.41	ASB Shield Building Roof Area
10471	1008	1014	134.25	CIS Operating Deck
9007	1000	1000	100	CIS at Reactor Vessel Support Elevation
11224	1000	1000	224	SCV Near Polar Crane

*Note: X=Y=1000 ft at center of ASB and SCV

The results at these six locations are compared with the CSDRS-based design envelops in Figure 12 through Figure 29. In these figures, X denotes plant North, Y denotes plant West and, Z denotes vertical direction.

For a point of reference, the comparisons also include the original AP1000 CSDRS broadened envelope ISRS to aid in understanding the differences in the revised ISRS comparison.

As shown in these figures, the “design envelope” exceeds the site specific response motions basically over the entire range of frequencies and by a large margin. This margin is particularly large at the zero period acceleration level indicating a large margin for seismic member forces. At a very limited frequency range, small exceedances beyond the design envelops are observed. The exceedance at about 0.55 Hz is consistent with the previous two-dimensional SSI results and has no design consequence since there are no structural members at this frequency.

7. Conclusion

The results of the three dimensional SSI analysis of a refined AP1000 NI model at the Vogtle site show a large margin against the design envelops. This study confirms the applicability of the AP1000 design to the Vogtle site.

8. Reference

1. AP1000 Standard Combined License Technical Report: Extension of Nuclear Island Seismic Analyses to Soil Sites; APP-GW-S2R-010, Revision 4, March 2010, Docket No. 52-006, Westinghouse letter dated April 21, 2010 (DCP_NRC_002855).

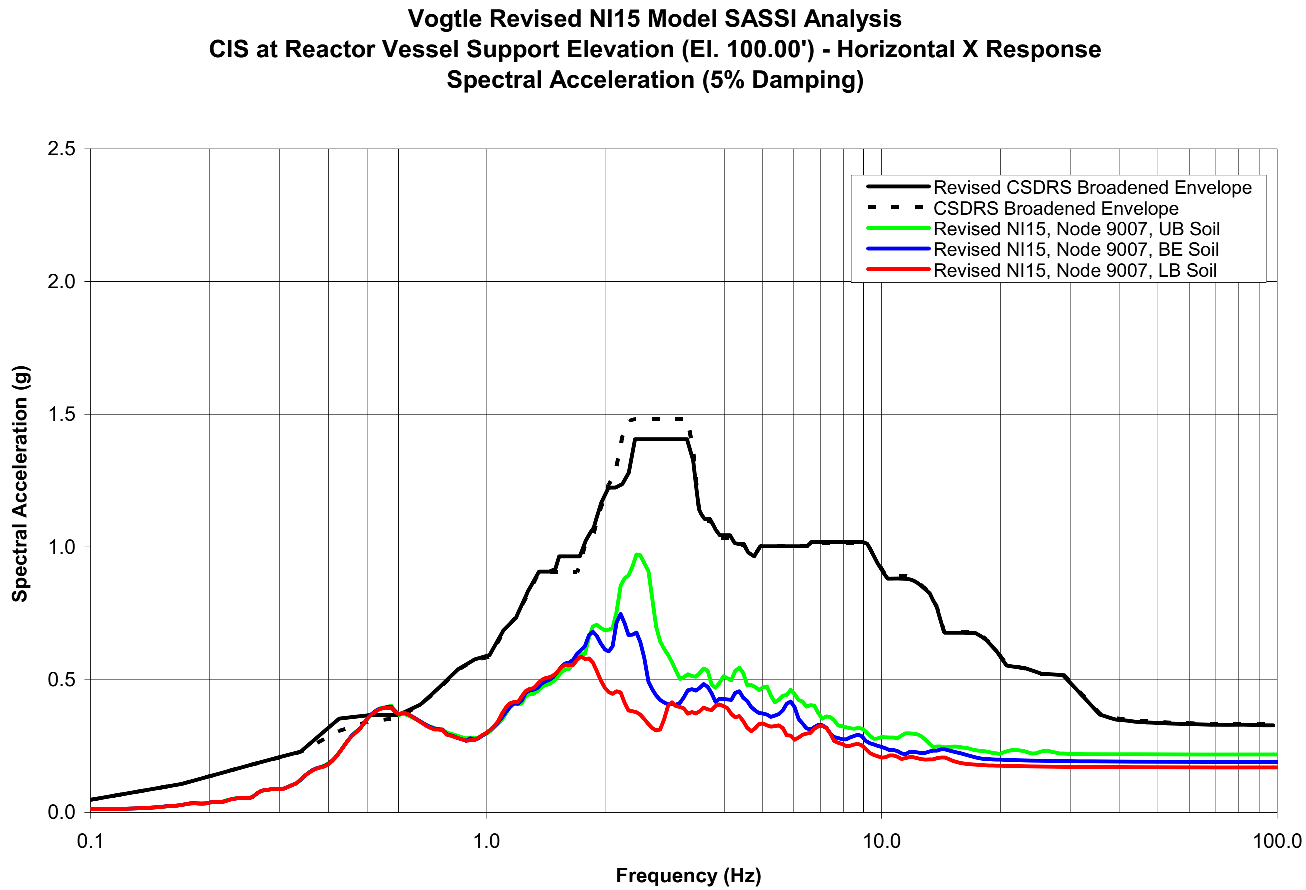

Figure 12 Horizontal X Response Spectra at CIS at Reactor Vessel Support Elevation (El. 100.00 ft, Node 9007)

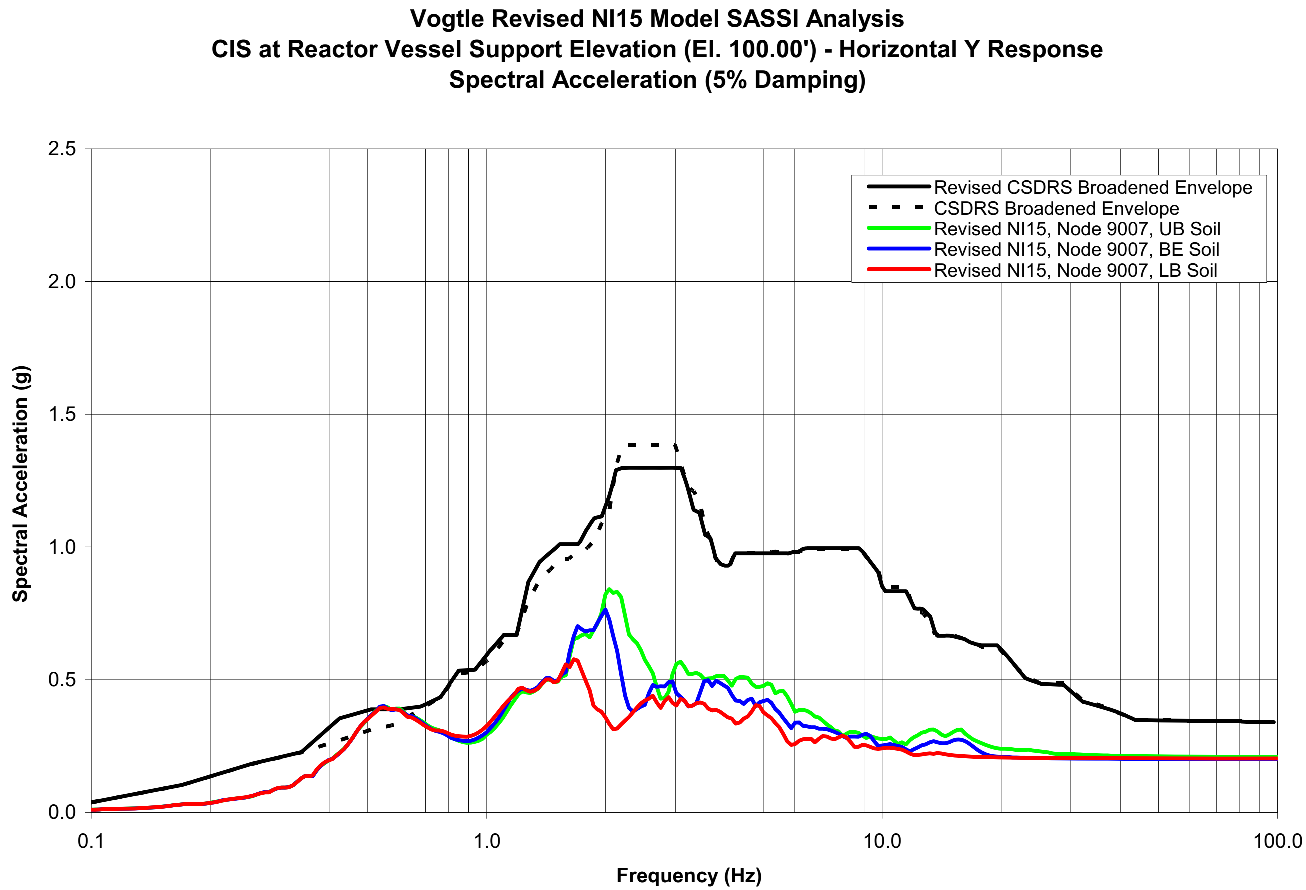

Figure 13 Horizontal Y Response Spectra at CIS at Reactor Vessel Support Elevation (El. 100.00 ft, Node 9007)

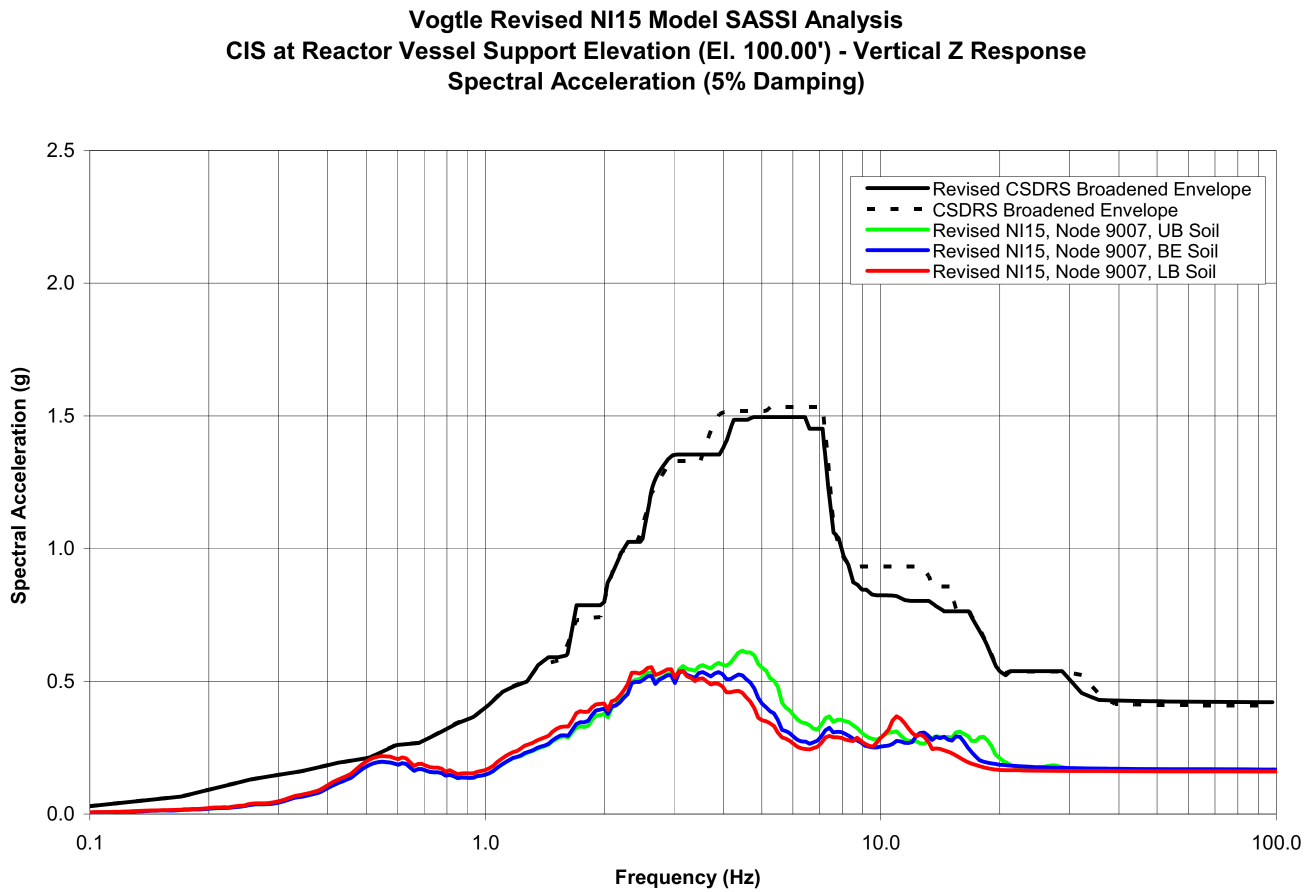

Figure 14 Vertical Z Response Spectra at CIS at Reactor Vessel Support Elevation (El. 100.00 ft, Node 9007)

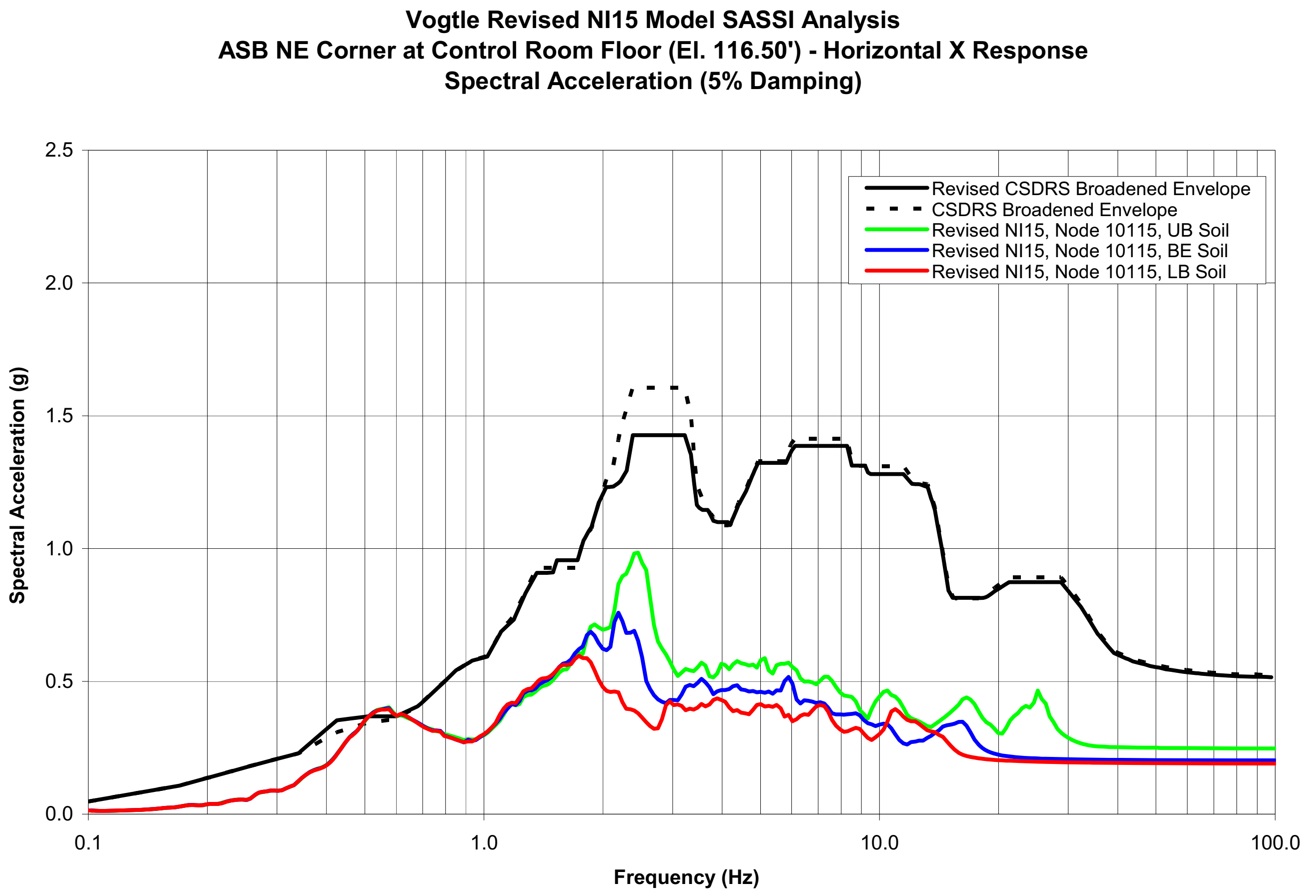

Figure 15 Horizontal X Response Spectra at ASB NE Corner at Control Room Floor (El. 116.50 ft, Node 10115)

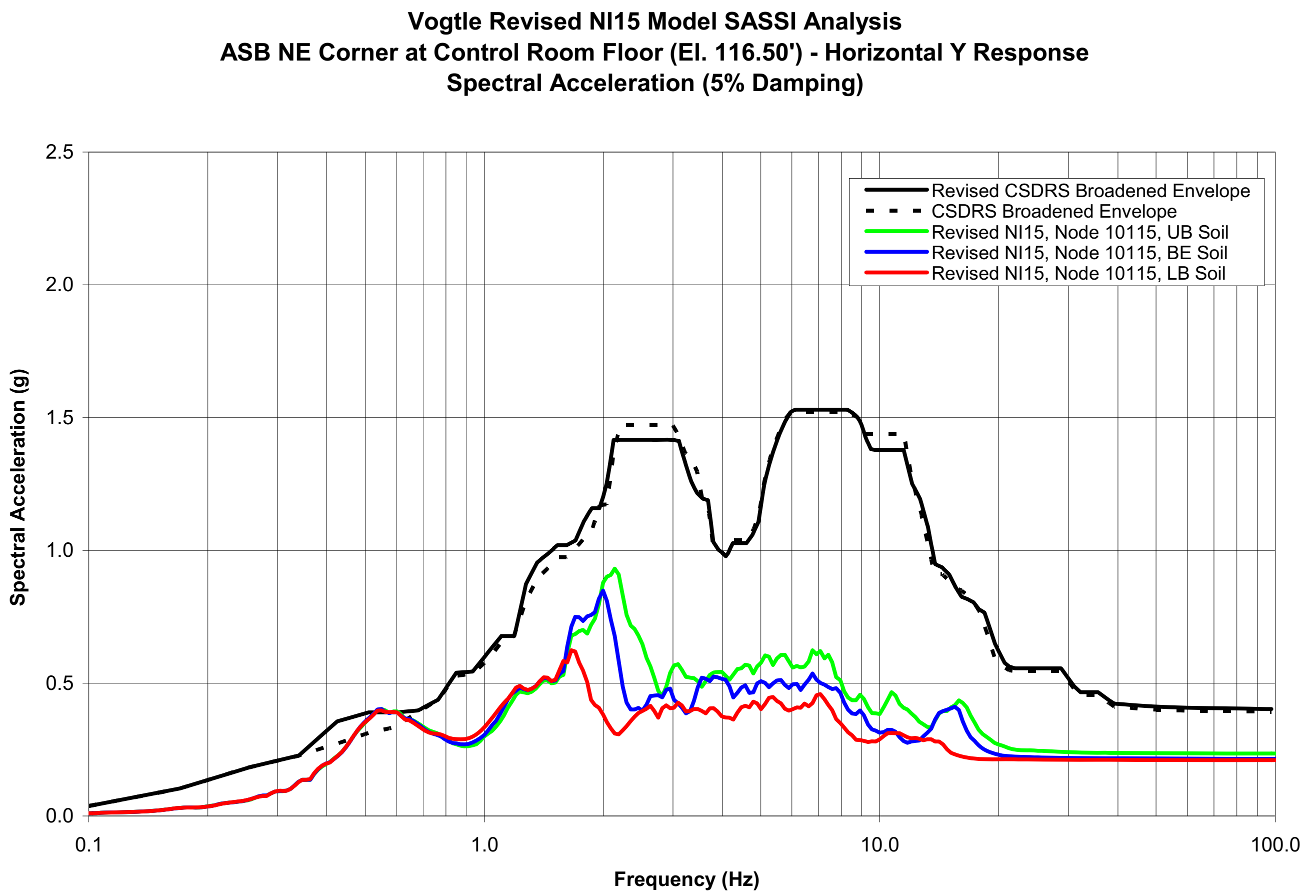

Figure 16 Horizontal Y Response Spectra at ASB NE Corner at Control Room Floor (El. 116.50 ft, Node 10115)

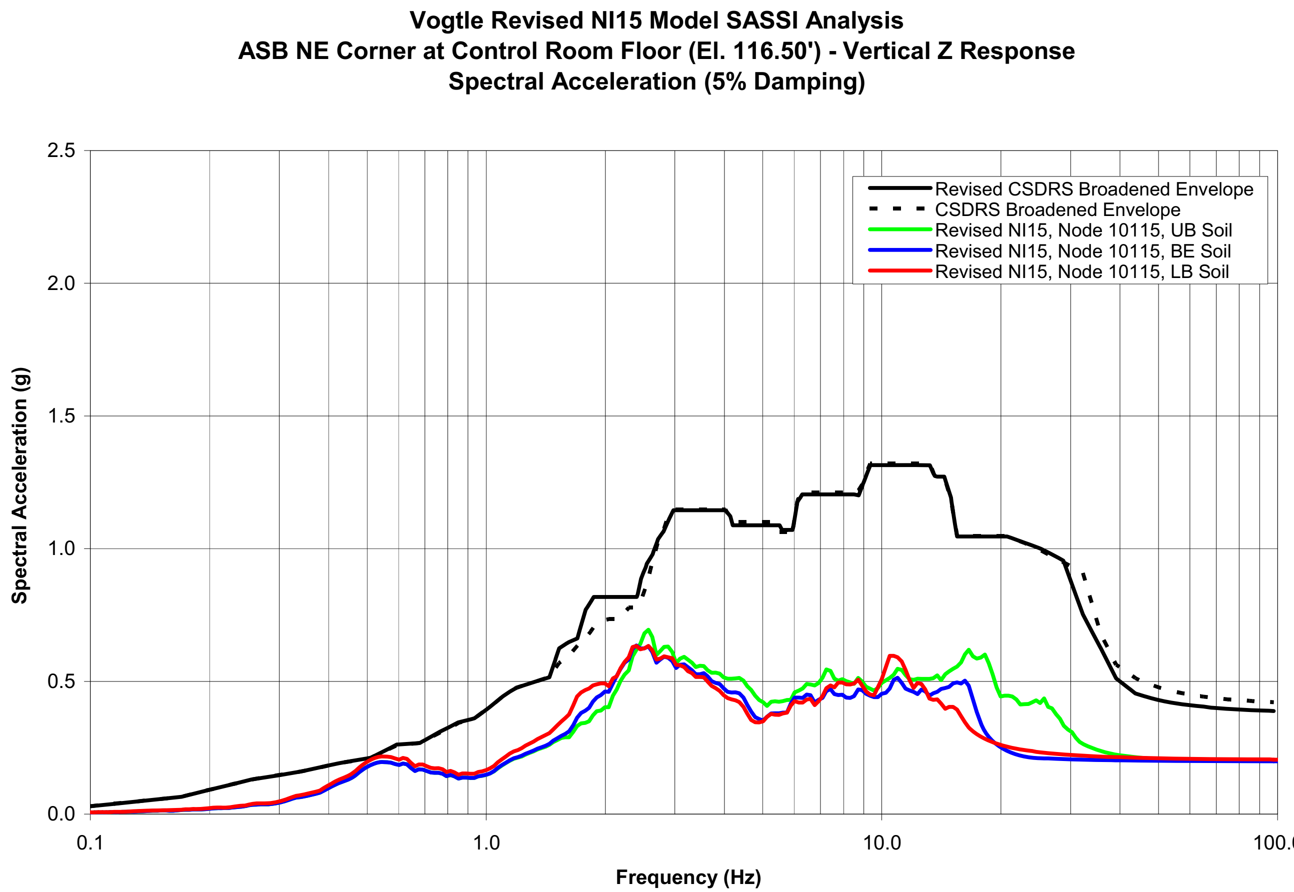

Figure 17 Vertical Z Response Spectra at ASB NE Corner at Control Room Floor (El. 116.50 ft, Node 10115)

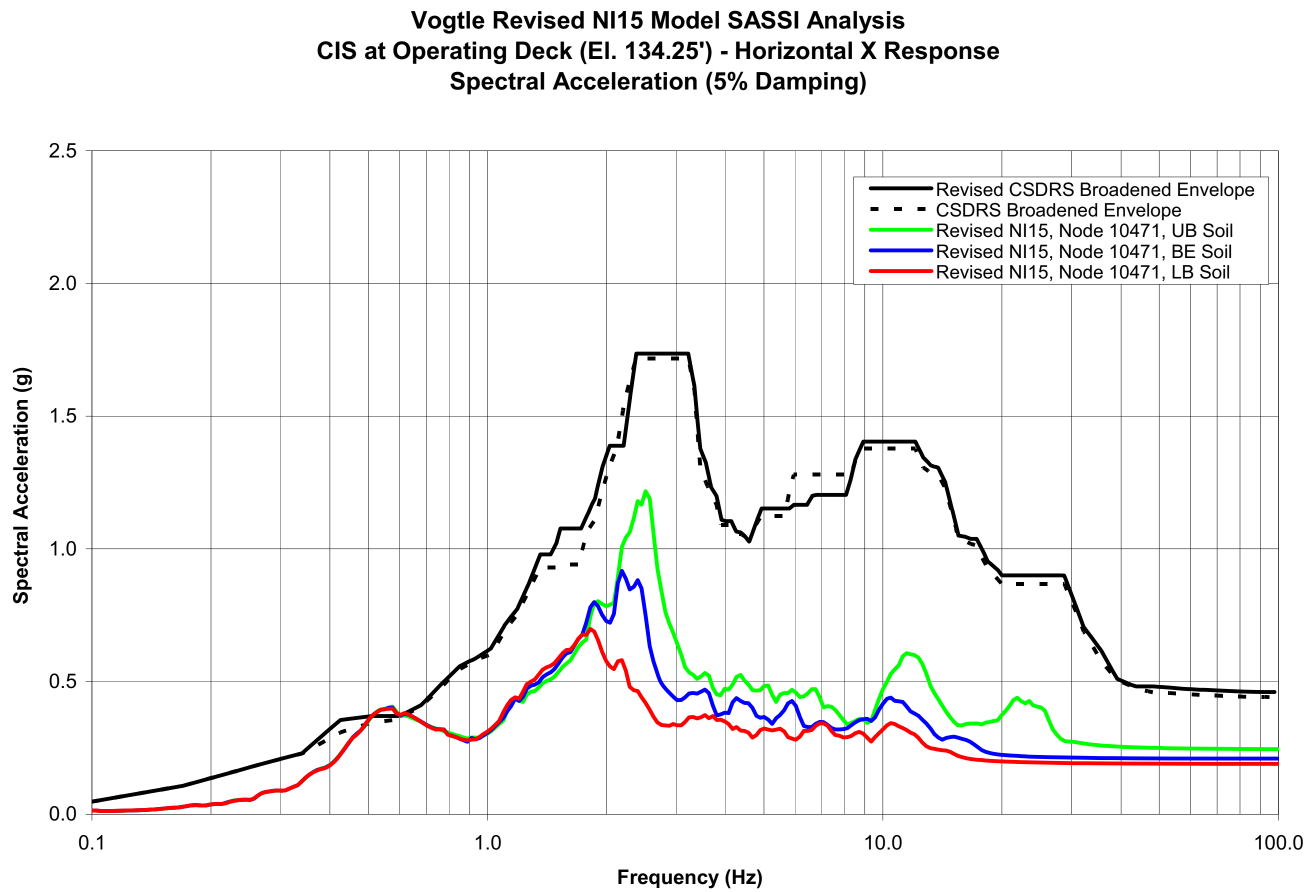

Figure 18 Horizontal X Response Spectra at CIS at Operating Deck (El. 134.25 ft, Node 10471)

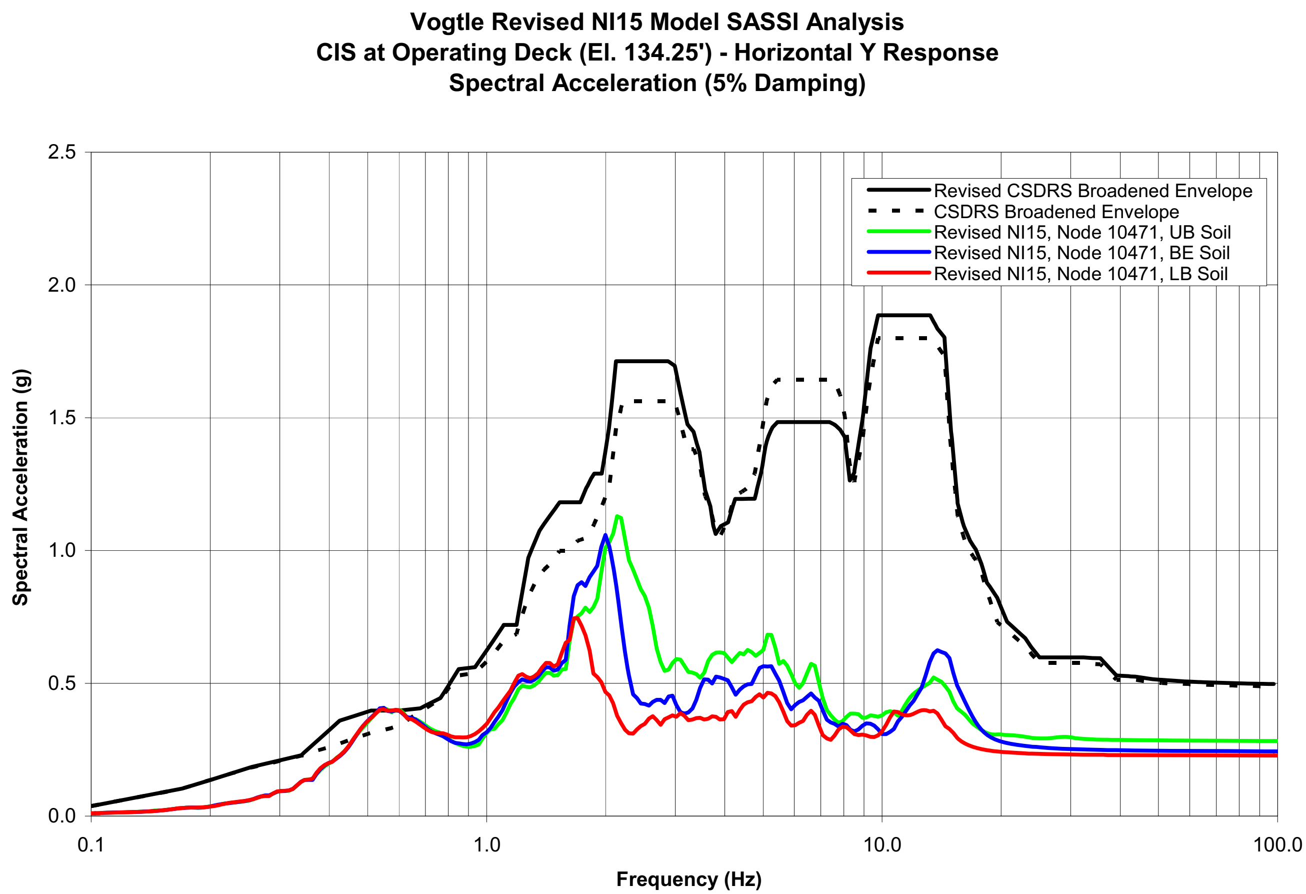

Figure 19 Horizontal Y Response Spectra at CIS at Operating Deck (El. 134.25 ft, Node 10471)

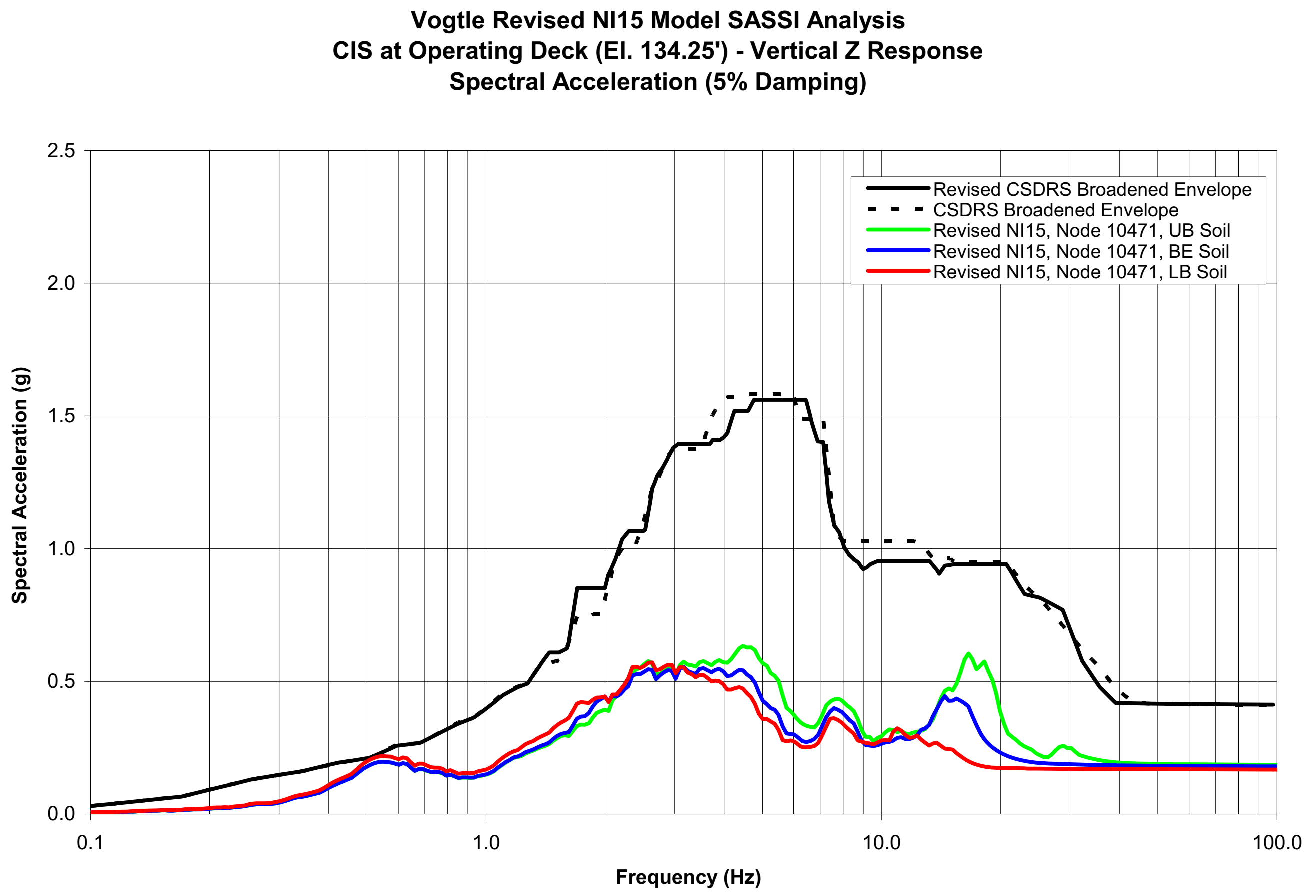

Figure 20 Vertical Z Response Spectra at CIS at Operating Deck (El. 134.25 ft, Node 10471)

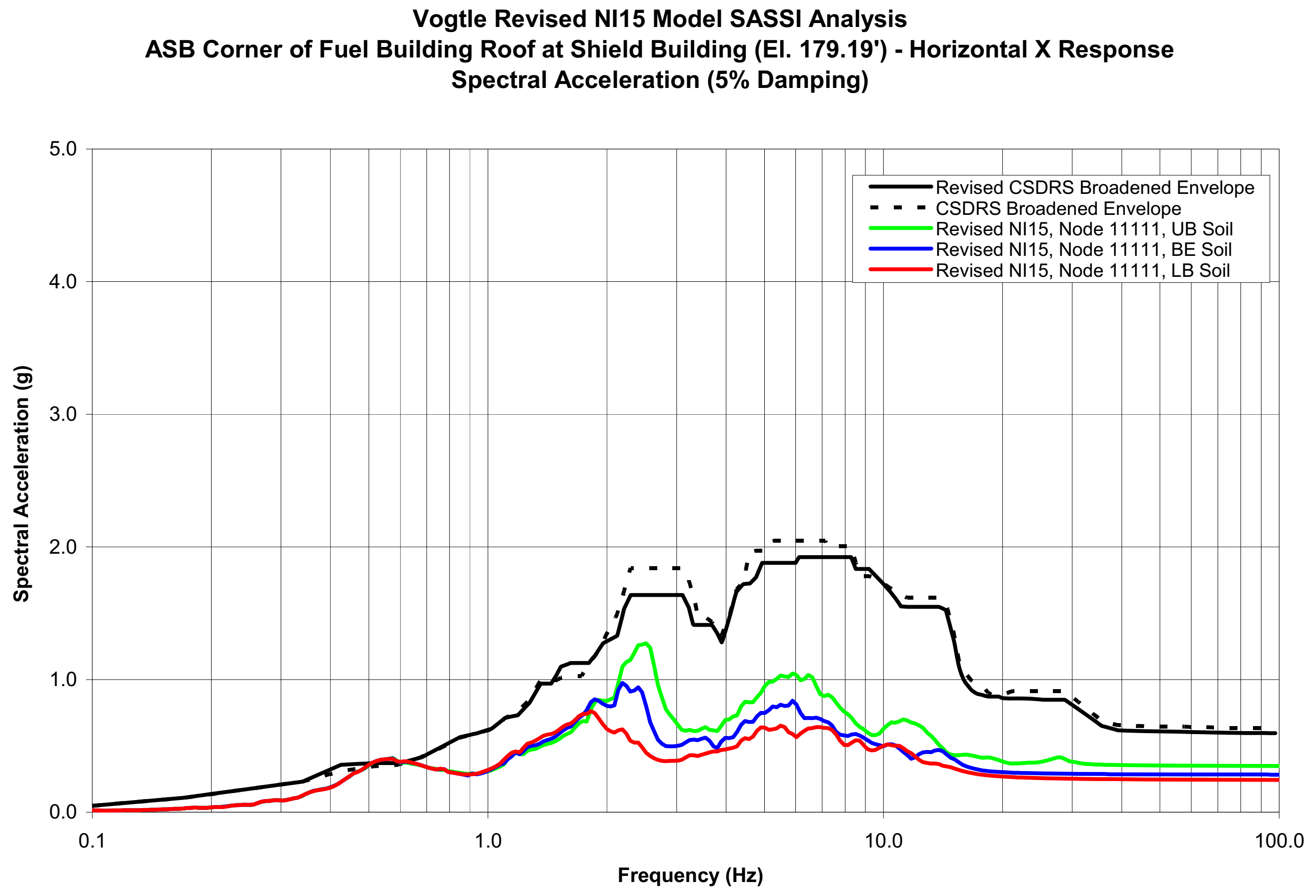

Figure 21 Horizontal X Response Spectra at ASB Corner of Fuel Building Roof at Shield Building (El. 179.19 ft, Node 11111)

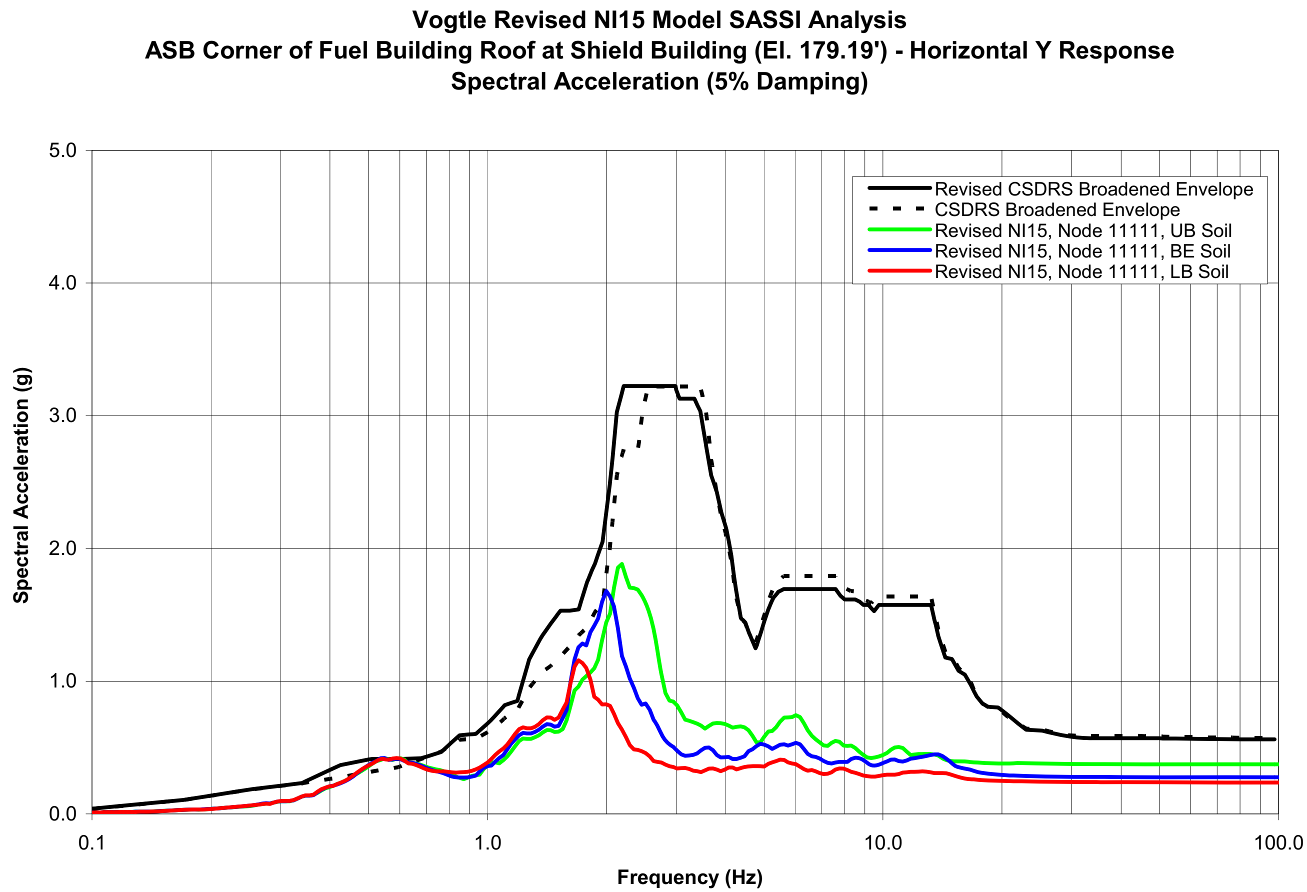

Figure 22 Horizontal Y Response Spectra at ASB Corner of Fuel Building Roof at Shield Building (El. 179.19 ft, Node 11111)

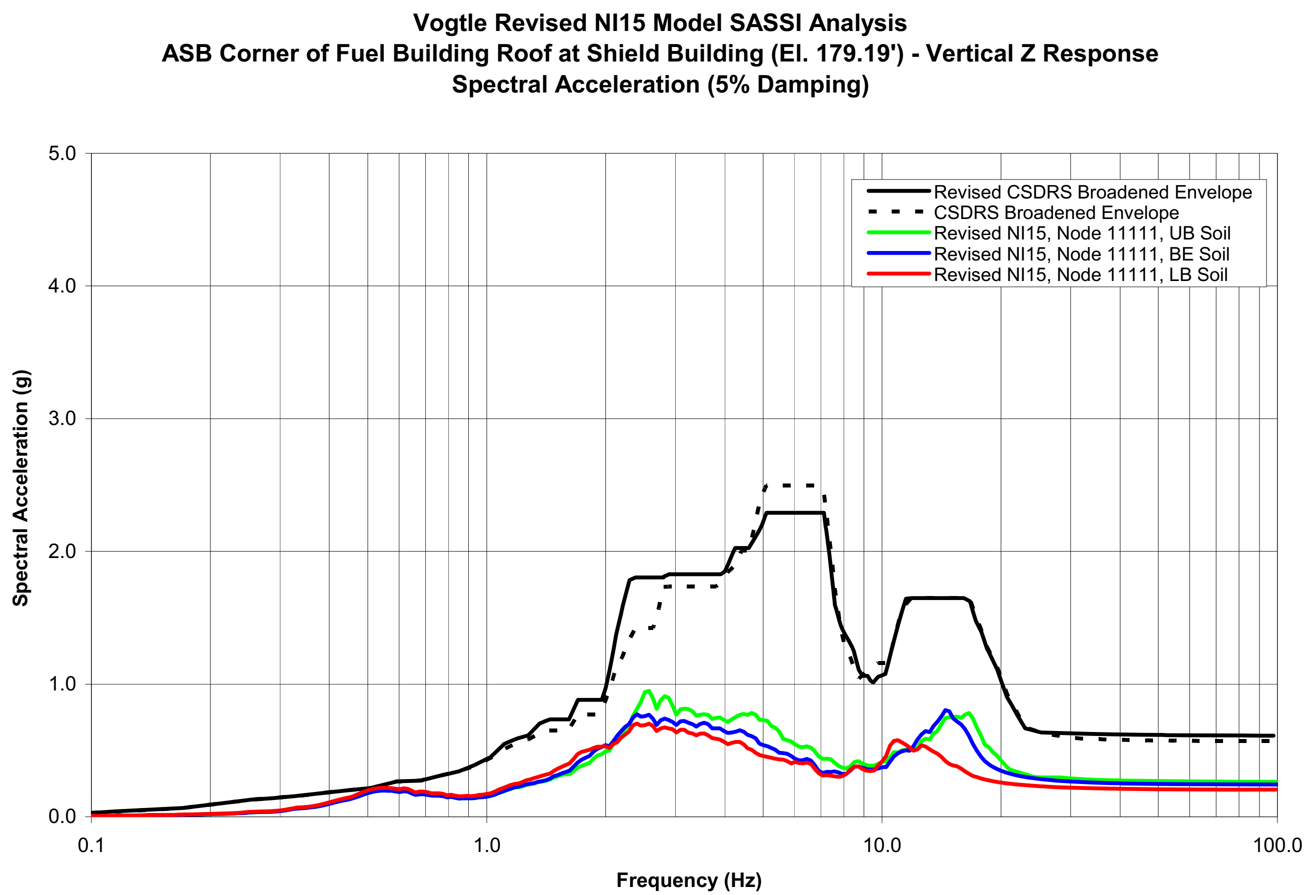

Figure 23 Vertical Z Response Spectra at ASB Corner of Fuel Building Roof at Shield Building (El. 179.19 ft, Node 11111)

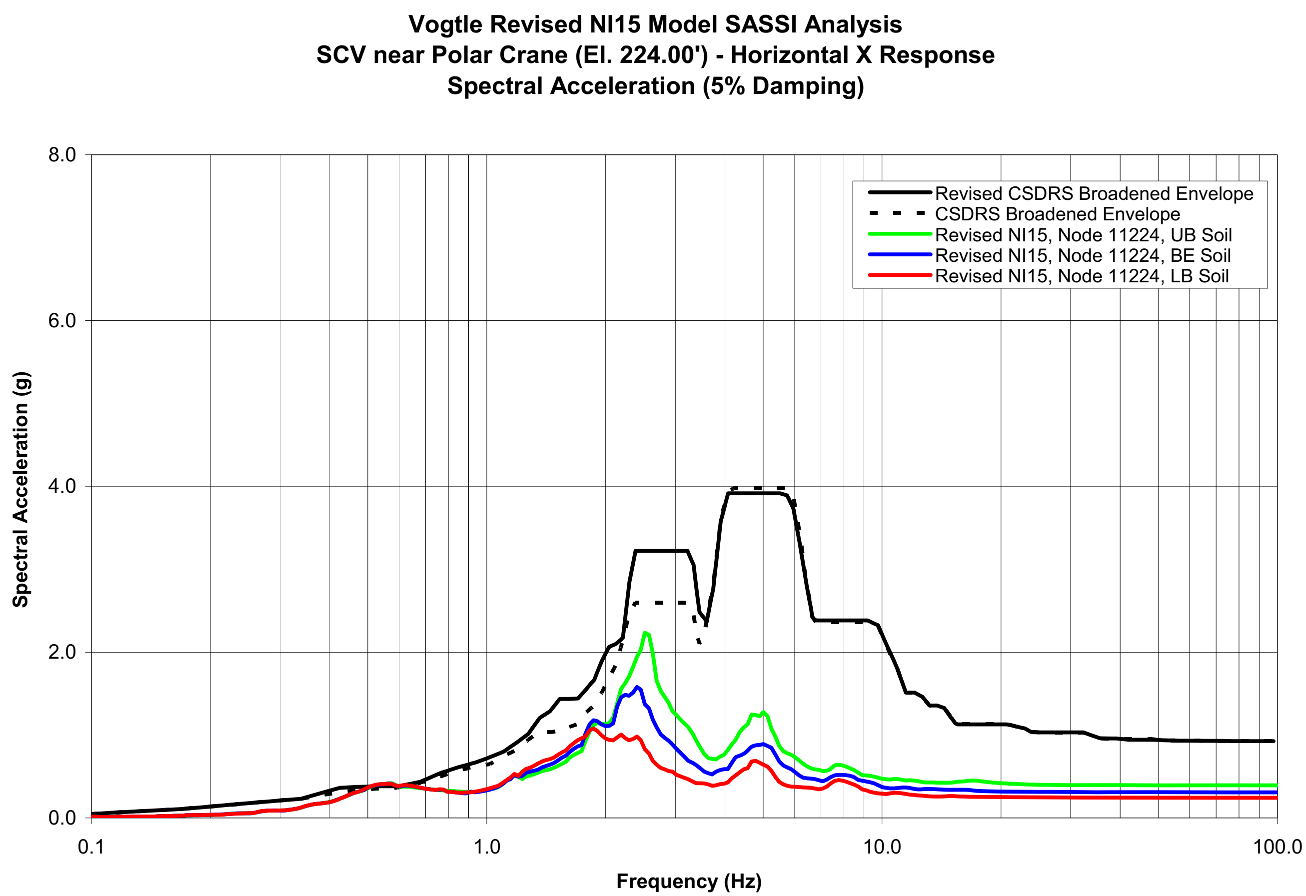

Figure 24 Horizontal X Response Spectra at SCV near Polar Crane (El. 224.00 ft, Node 11224)

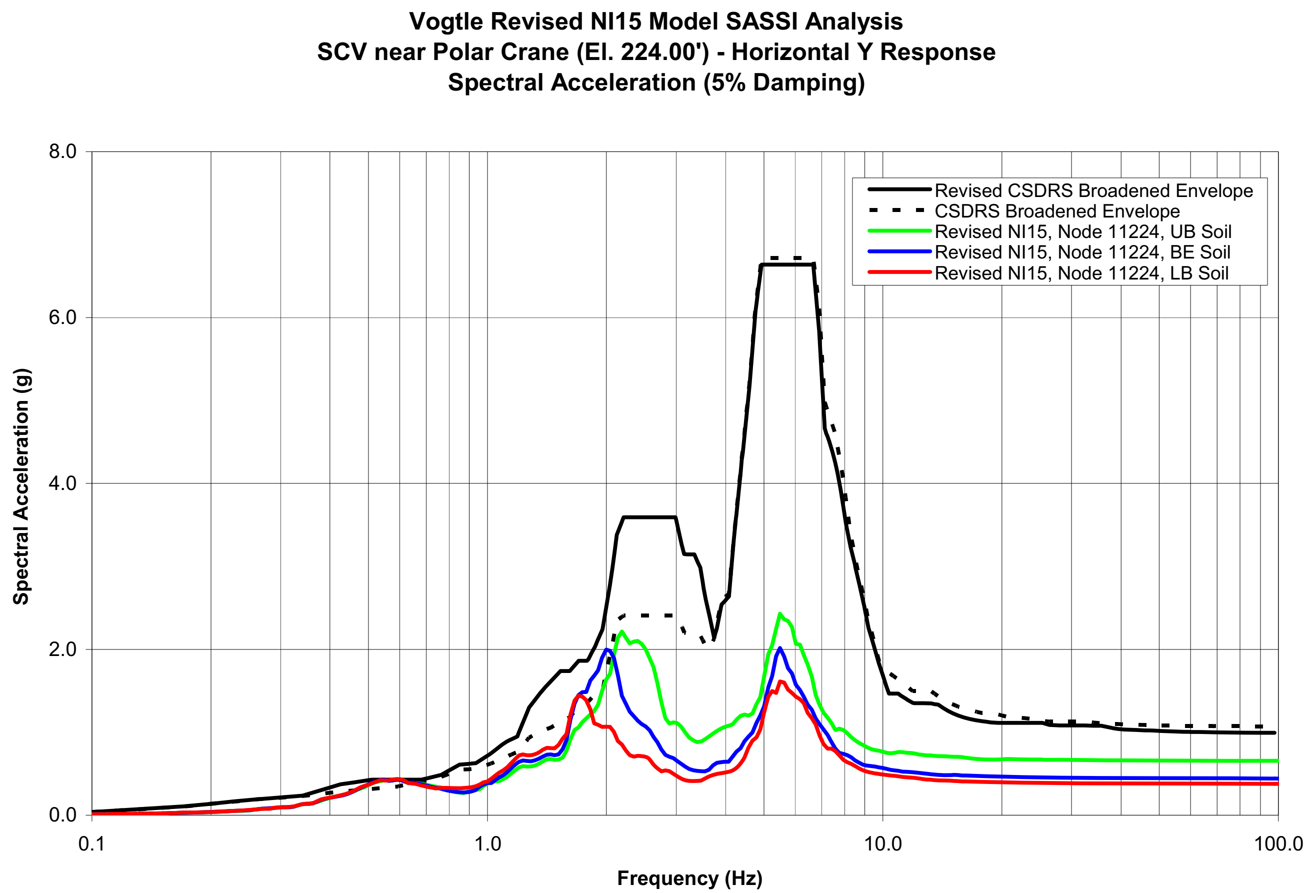

Figure 25 Horizontal Y Response Spectra at SCV near Polar Crane (El. 224.00 ft, Node 11224)

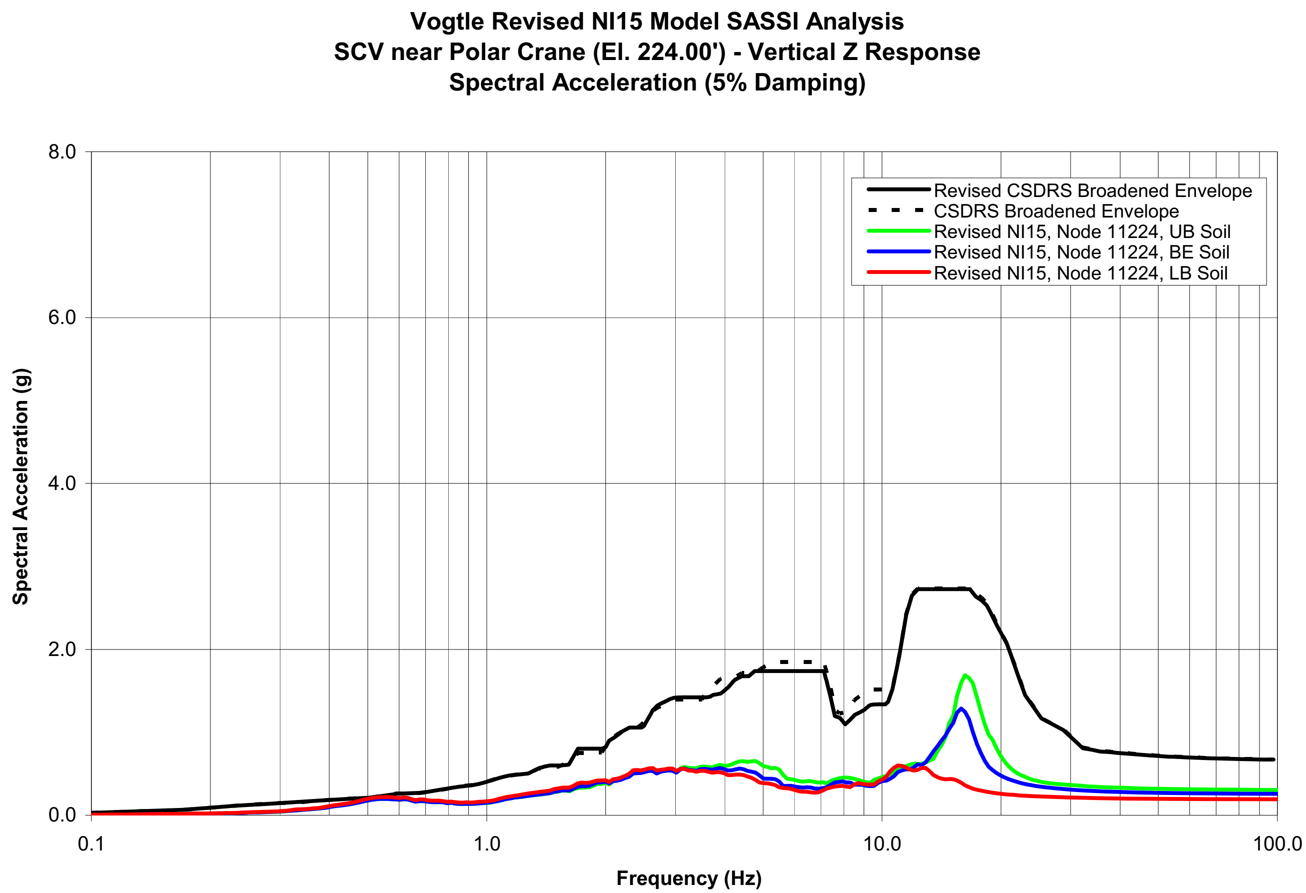

Figure 26 Vertical Z Response Spectra at SCV near Polar Crane (El. 224.00 ft, Node 11224)

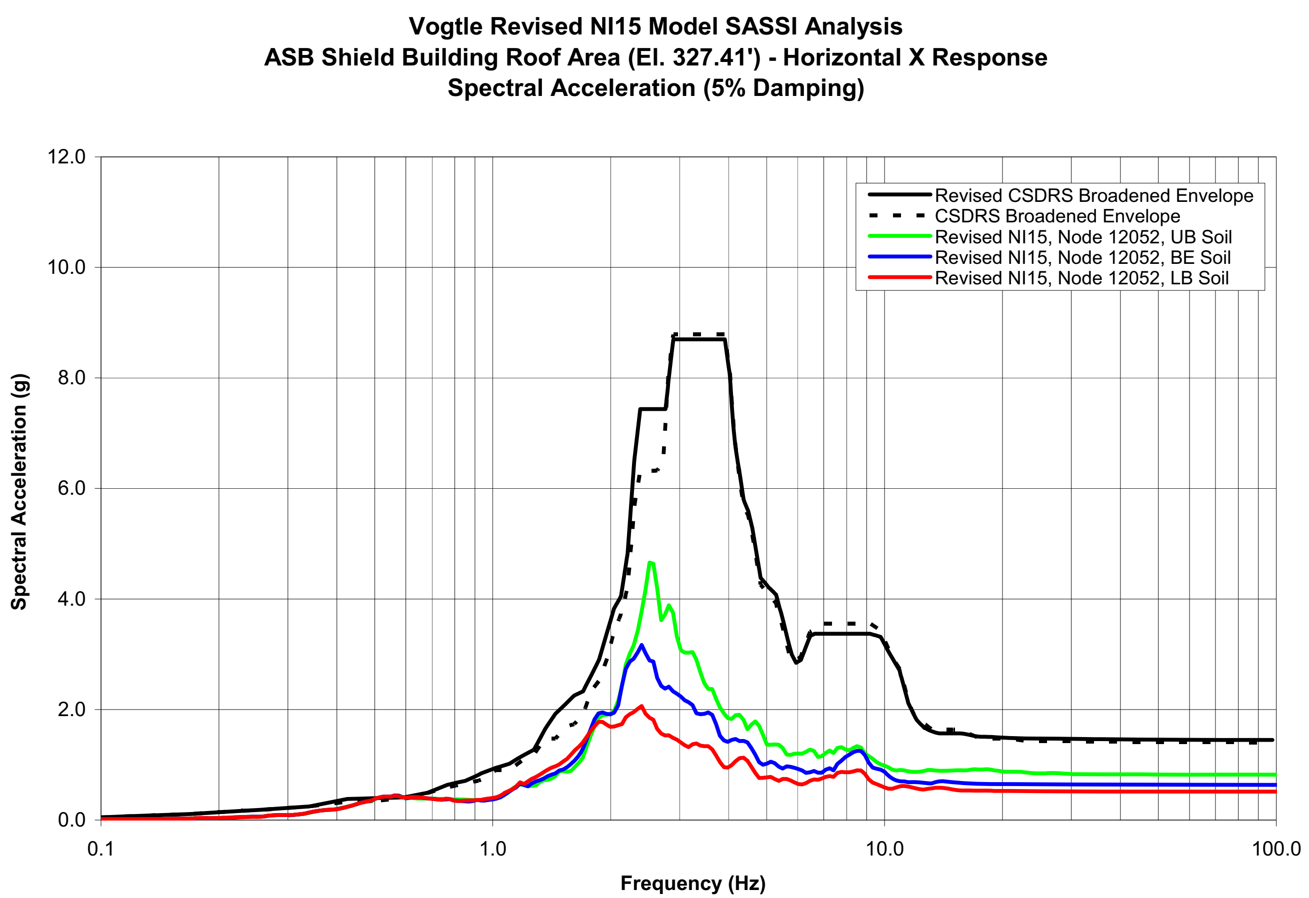

Figure 27 Horizontal X Response Spectra at ASB Shield Building Roof Area (El. 327.41 ft, Node 12052)

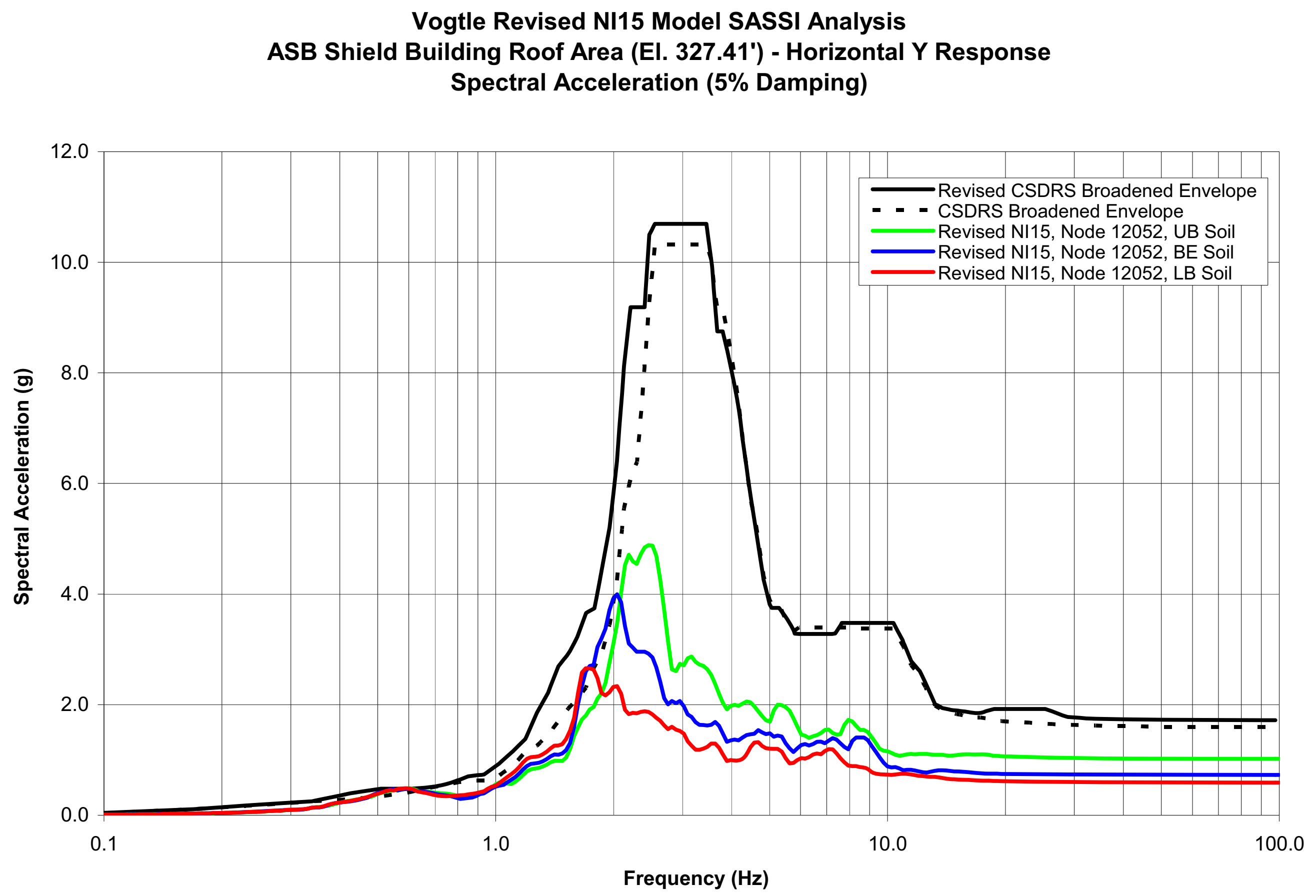

Figure 28 Horizontal Y Response Spectra at ASB Shield Building Roof Area (El. 327.41 ft, Node 12052)

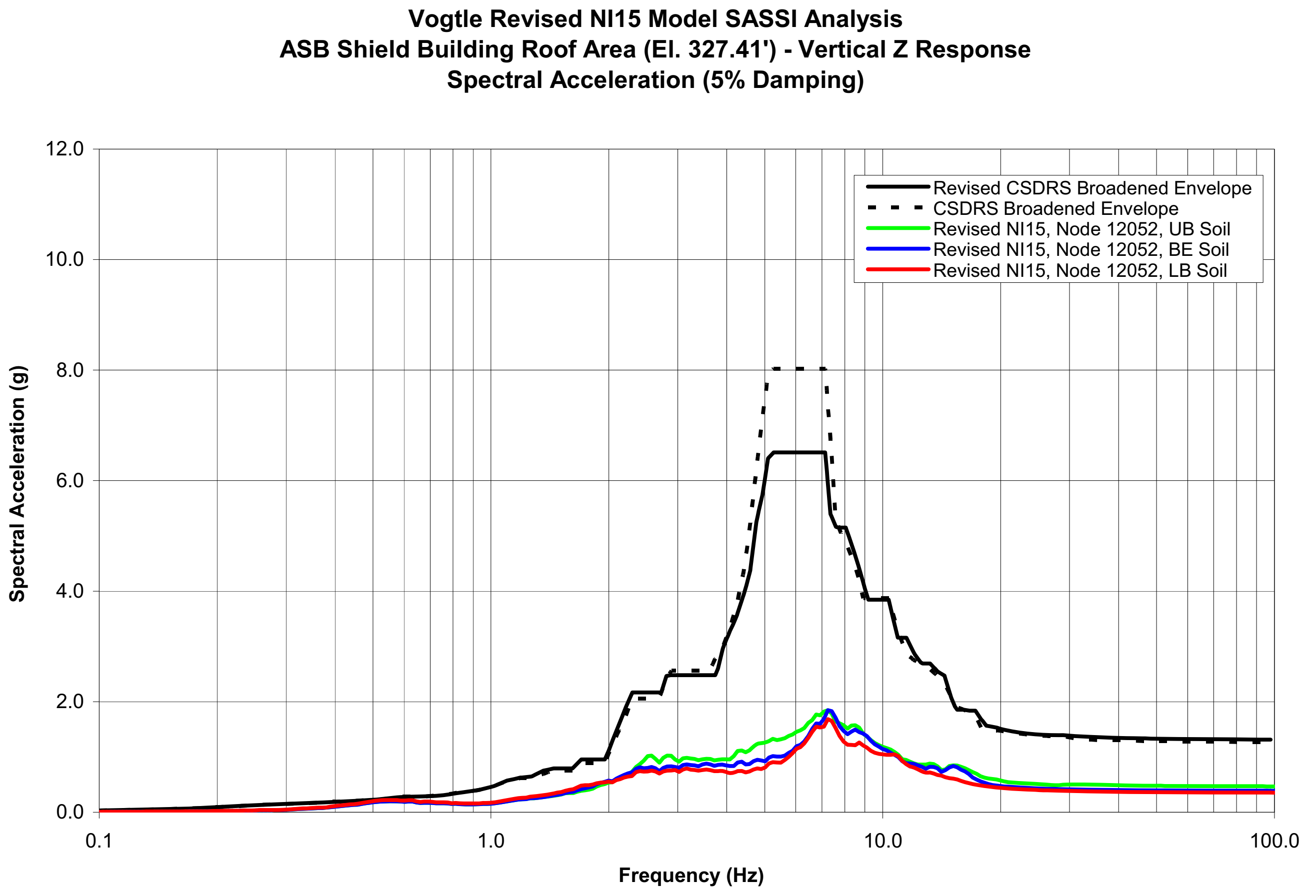

Figure 29 Vertical Z Response Spectra at ASB Shield Building Roof Area (El. 327.41 ft, Node 12052)

APPENDIX 3H
AUXILIARY AND SHIELD BUILDING CRITICAL SECTIONS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 3I
EVALUATION FOR HIGH FREQUENCY SEISMIC INPUT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 4

REACTOR

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CHAPTER 4
REACTOR

4.1 SUMMARY DESCRIPTION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

4.2 FUEL SYSTEM DESIGN

This section of the referenced DCD is incorporated by reference with no departures or supplements.

4.3 NUCLEAR DESIGN

This section of the referenced DCD is incorporated by reference with no departures or supplements.

4.4 THERMAL AND HYDRAULIC DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

4.4.7 COMBINED LICENSE INFORMATION

Replace the paragraph in DCD Subsection 4.4.7.2 with the following:

STD COL 4.4-2 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 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.

4.5 REACTOR MATERIALS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 5

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

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CHAPTER 5
REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

5.2.1.1 Compliance with 10 CFR 50.55a

Add the following text after the second sentence of the second paragraph of DCD Subsection 5.2.1.1.

STD COL 5.2-1 If a later Code edition/addenda than the Design Certification Code edition/addenda is used by the material and/or component supplier, then a code reconciliation to determine acceptability is performed as required by the ASME Code, Section III, NCA-1140. The later Code edition/addenda must be authorized in 10 CFR 50.55a or in a specific authorization as provided in 50.55a(a)(3). Code Cases to be used in design and construction are identified in the DCD; additional Code Cases for design and construction beyond those for the design certification are not required.

Inservice inspection of the reactor coolant pressure boundary is conducted in accordance with the applicable edition and addenda of the ASME Boiler and Pressure Vessel Code Section XI, as described in Subsection 5.2.4. Inservice testing of the reactor coolant pressure boundary components is in accordance with the edition and addenda of the ASME OM Code as discussed in Subsection 3.9.6 for pumps and valves, and as discussed in Subsection 3.9.3.4.4 for dynamic restraints.

5.2.3.2.1 Chemistry of Reactor Coolant

Add the following text to the end of DCD Subsection 5.2.3.2.1.

STD SUP 5.2-1 The water chemistry program is based on industry guidelines as described in EPRI TR-1002884, “Pressurized Water Reactor Primary Water Chemistry” (Reference 201). 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 to within the specified range.

Chemistry procedures will provide guidance for the sampling and monitoring of primary coolant properties.

5.2.4 INSERVICE INSPECTION AND TESTING OF CLASS 1 COMPONENTS

Add the following after the first paragraph in DCD Subsection 5.2.4:

STD COL 5.2-2 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 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 ASME 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).

5.2.4.1 System Boundary Subject to Inspection

Add the following at the end of DCD Subsection 5.2.4.1:

STD COL 5.2-2 The Class 1 system boundary for both preservice and inservice inspection programs and the system pressure test program includes those items within the Class 1 and Quality Group A (Equipment Class A per DCD Subsection 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);
- Portions of the Incore Instrumentation System (IIS);
- Portions of the Passive Core Cooling System (PXS);
- Portions of the Reactor Coolant System (RCS); and
- Portions of the Normal Residual Heat Removal System (RNS).

Those portions of the above systems within the Class 1 boundary are those items that are part of the reactor coolant pressure boundary as defined in Section 5.2.

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 reactor 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 in series (both closed, both open, or one closed and the 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 description of portions of systems excluded from the RCPB does not address Class 1 components exempt from inservice examinations under ASME Code Section XI rules. The Class 1 components exempt from inservice examinations are defined by ASME Section XI, IWB-1220, except as modified by 10 CFR 50.55a.

The inservice inspection program is augmented for reactor vessel top head inspections by use of the ASME Code Case N-729-1, “Alternative Examination Requirements for Pressurized-Water Reactor (PWR) Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, “as modified by the conditions specified in 10 CFR 50.55a(g)(6)(ii)(D).

Boric acid corrosion control procedures require inspection of the reactor coolant pressure boundary subject to leakage that can cause boric acid corrosion of the reactor coolant pressure boundary materials. The procedures determine the principal locations where leaks can cause degradation of the primary pressure boundary by boric acid corrosion. Potential paths of the leaking coolant are established. The boric acid corrosion control procedures also contain methods for conducting examinations and performing engineering evaluations to establish the impact on the reactor coolant pressure boundary when leakage is located.

The boric acid corrosion control procedures consist of:

1. Visual inspections of component surfaces that are potentially exposed to borated water leakage.

2. Discovery of leak path and removal of boric acid residue.

3. Assessment of the corrosion.

4. Follow-up inspection for adequacy of corrective actions, as appropriate.

Add the following text at the end of DCD Subsection 5.2.4.1:

STD SUP 5.2-2 The inservice inspection program, along with the boric acid corrosion control procedures, provides guidance for inspecting the integrity of bolting and threaded fasteners.

STD COL 5.3-7 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.

5.2.4.3 Examination Techniques and Procedures

Add the following at the end of DCD Subsection 5.2.4.3:

5.2.4.3.1 Examination Methods

Ultrasonic Examination of the Reactor Vessel

STD COL 5.2-2 Ultrasonic examination for the RPV is conducted in accordance with the ASME Code, Section XI. The design of the RPV considered the requirements of the ASME Code Section XI with regard to performance of preservice inspection. For the required preservice examinations, the reactor vessel meets the acceptance standards of Section XI, IWB-3510. The RPV shell welds are designed for 100% accessibility for both preservice and inservice inspection. RPV shell welds may be examined from the inside or outside diameter surfaces (or a combination of those techniques) using automated ultrasonic examination equipment. The RPV nozzle-to-shell welds are 100% 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.

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).

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 meet the requirements of IWA-5240.

Where direct visual 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 and liquid penetrant 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 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.

Volumetric Ultrasonic Direct Examination

Volumetric ultrasonic direct examination is performed in accordance with ASME Section XI, IWA-2232, which references mandatory Appendix I.

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 method. This provision allows for the use of newly developed examination 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, is in compliance with the provisions of 10 CFR 50.55a.

5.2.4.4 Inspection Intervals

Add the following after the second sentence of the first paragraph of DCD Subsection 5.2.4.4:

STD COL 5.2-2 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. Period one comprises the first three years of the interval, period two comprises the next four years of the interval, and period three comprises the remaining three years of the inspection interval. Each period can be extended for 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.

5.2.4.5 Examination Categories and Requirements

Add the following after the first sentence of DCD Subsection 5.2.4.5:

STD COL 5.2-2 Class 1 piping supports will be examined in accordance with ASME Section XI, IWF-2500.

Preservice examinations required by design specification 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.

Add the following after the last sentence of DCD Subsection 5.2.4.5:

The preservice examination is performed once in accordance with ASME 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 as 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.

5.2.4.6 Evaluation of Examination Results

Add the following at the end of DCD Subsection 5.2.4.6:

STD COL 5.2-2 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.

STD COL 5.2-2 Add Subsections 5.2.4.8, 5.2.4.9, and 5.2.4.10 after the last paragraph of DCD Subsection 5.2.4.7:

5.2.4.8 Relief Requests

The specific areas where the applicable ASME Code requirements cannot be met are identified after the initial examinations are performed. Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(a)(3) or 50.55a(g)(5).

The relief requests include appropriate justifications and proposed alternative inspection methods.

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.

5.2.4.10 Program Implementation

The milestones for preservice and inservice inspection program implementation are identified in Table 13.4-201.

Add the following new subsection following DCD Subsection 5.2.5.3.4.

5.2.5.3.5 Response to Reactor Coolant System Leakage

STD COL 5.2-3 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 results 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.

5.2.6 COMBINED LICENSE INFORMATION ITEMS

5.2.6.1 ASME Code and Addenda

STD COL 5.2-1 This COL Item is addressed in Subsection 5.2.1.1.

5.2.6.2 Plant-Specific Inspection Program

STD COL 5.2-2 This COL Item is addressed in Subsections 5.2.4, 5.2.4.1, 5.2.4.3.1, 5.2.4.3.2, 5.2.4.4, 5.2.4.5, 5.2.4.6, 5.2.4.8, 5.2.4.9, and 5.2.4.10.

5.2.6.3 Response to Unidentified Reactor Coolant System Leakage Inside Containment

STD COL 5.2-3 This COL item is addressed in Subsection 5.2.5.3.5.

5.2.7 REFERENCES

Add the following information at the end of DCD Subsection 5.2.7.

201. EPRI, "Pressurized Water Reactor Primary Water Chemistry Guidelines," EPRI TR-1002884, Revision 5, October 2003.

5.3 REACTOR VESSEL

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

5.3.2.6 Material Surveillance

Add the following information between the first and second paragraphs of DCD Subsection 5.3.2.6.

STD COL 5.3-2 Surveillance test materials are prepared from the actual materials used in fabricating the beltline region of the reactor vessel. Records are maintained of the chemical analyses, fabrication history, mechanical properties and other essential variables pertinent to the fabrication process of the shell forging and weld metal from which the surveillance test materials are prepared. The test materials are processed so that they are representative of the material in the completed reactor vessel.

Three metallurgically different materials prepared from sections of reactor vessel shell forging are used for test specimens. These include base metal, weld metal and heat affected zone (HAZ) material.

Base metal test material is manufactured from a section of ring forging, either the intermediate shell course, the lower shell course, or the transition ring of the reactor pressure vessel. Selection is based on an evaluation of initial toughness (characterized by the reference temperature (RT_{NDT}) and Upper Shelf Energy (USE)), and the predicted effect of chemical composition (nickel and residual copper) and neutron fluence on the toughness (RT_{NDT} shift and decrease in USE) during reactor operation. The ring forging with the highest predicted adjusted RT_{NDT} temperature (initial RT_{NDT} plus RT_{NDT} shift) or that with USE predicted to approach close to the minimum limit of 50 ft-lb at end-of-license (EOL) is selected as the surveillance base metal test material. The means for measuring initial toughness and for predicting irradiation induced toughness changes is consistent with applicable procedures in force at the time the material is being selected. The section of shell forging used for the base metal test block is adjacent to the test material used for fracture toughness tests.

Weld metal and HAZ test material is produced by welding together sections of the forgings from the beltline of the reactor vessel. The HAZ test material is manufactured from a section of the same shell course forging used for base metal test material. The sections of shell course forging used for weld metal and HAZ test material are adjacent to the test material used for fracture toughness tests. The heat of wire or rod and lot of flux are from the same heat and lot used in making the beltline region welds. Welding parameters duplicate those used for the beltline region welds. The procedures for inspection of the reactor vessel welds are followed for the inspection of the welds in test materials. The surveillance weld and HAZ material are heat-treated to metallurgical conditions which are representative of the final metallurgical conditions of similar materials in the completed reactor vessel.

Test Specimens are marked to identify the type of materials and the orientation with respect to the test materials. Drawings specify the identification system to be used and include plant identification, type of material, orientation of specimen and sequential number.

Baseline test specimens are provided for establishing the baseline (unirradiated) properties of the reactor vessel materials. The data from tests of these specimens provides the basis for determining the radiation induced property changes of the reactor vessel materials.

Drop weight test specimens of each of base metal, weld metal, and HAZ metal are provided for establishing the nil-ductility transition temperature (NDTT) of the unirradiated surveillance materials. These data form the basis for RT_{NDT} determination from which subsequent radiation induced changes are determined.

Standard Charpy impact test specimens each of base metal (longitudinal (tangential) and transverse (axial)), weld metal, and HAZ material are provided for developing a Charpy impact energy transition curve from fully brittle to fully ductile behavior for defining specific index temperatures for these materials. These data, together with the drop weight NDTT, are used to establish an RT_{NDT} for each material.

Tensile test specimens each of base metal (longitudinal (tangential) and transverse (axial)), weld metal, and HAZ metal are provided to permit a sufficient number of tests for accurately establishing the tensile properties for these materials at a minimum of three test temperatures (e.g., ambient, operating and one intermediate temperature) to define the strength of the material.

The above described test specimens are to be used for determining changes in the strength and toughness of the surveillance materials resulting from neutron irradiation. Sufficient Charpy impact, compact tension and tensile test specimens are provided for establishing the changes in the properties of the surveillance materials over the lifetime of the reactor vessel. The type, quantity, and storage conditions (e.g., surveillance capsules backfilled with inert gas) of test specimens meet or exceed the minimum requirements of ASTM E-185.

Reactor materials do not begin to be affected by neutron fluence until the reactor begins critical operation. Table 13.4-201 provides milestones for reactor vessel material surveillance program implementation.

Add the following subsection after DCD Subsection 5.3.2.6.2.2.

5.3.2.6.3 Report of Test Results

STD COL 5.3-2 A summary technical report for each capsule withdrawn with the test results is submitted, as specified in 10 CFR 50.4, within one year of the date of capsule withdrawal unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.

The report includes the data required by ASTM E185-82, as specified in paragraph III.B.1 of 10 CFR Part 50, Appendix H, and includes the results of the fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

If the test results indicate a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits, the expected date for submittal of the revised Technical Specification is provided with the report.

Add the following subsection after DCD Subsection 5.3.3.1.

5.3.3.2 Operating Procedures

STD SUP 5.3-1 Plant operating procedures are developed and maintained to prevent exceeding the pressure-temperature limits identified in reactor coolant system pressure and temperature limits report, as required by Technical Specification 5.6.6, during normal and abnormal operating conditions and system tests.

5.3.6 COMBINED LICENSE INFORMATION

5.3.6.1 Pressure-Temperature Limit Curves

Replace the text in DCD Subsection 5.3.6.1 with the following.

STD COL 5.3-1 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 the 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 the plant-specific pressure-temperature curves. The development of the plant-specific curves and evaluation of the setpoint pressure are required prior to fuel load.

5.3.6.2 Reactor Vessel Materials Surveillance Program

STD COL 5.3-2 This COL Item is addressed in Subsections 5.3.2.6 and 5.3.2.6.3.

5.3.6.4 Reactor Vessel Materials Properties Verification

Replace the text in DCD Subsection 5.3.6.4.1 with the following.

5.3.6.4.1 Reactor Vessel Materials Properties Verification

STD COL 5.3-4 The verification of plant-specific belt line material properties consistent with the requirements in DCD Subsection 5.3.3.1 and DCD Tables 5.3-1 and 5.3-3 will be completed prior to fuel load. The verification will include a pressurized thermal shock evaluation based on as procured reactor vessel material data and the projected neutron fluence for the plant design objective of 60 years. This evaluation report will be submitted for NRC staff review.

5.3.6.6 Quickloc Weld Build-up ISI

STD COL 5.3-7 This item is addressed in Subsection 5.2.4.1.

5.4 COMPONENT AND SUBSYSTEM DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

5.4.2.5 Steam Generator Inservice Inspection

Add the following information at the end of DCD Subsection 5.4.2.5.

STD COL 5.4-1 A steam generator tube surveillance program is implemented in accordance with the recommendations and guidance of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines" (Reference 201). A program for periodic monitoring of degradation of steam generator internals is also implemented in accordance with NEI 97-06. Applicable Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP) guidelines are followed as described in the NEI 97-06. The Programs are in compliance with applicable sections of ASME Section XI.

NEI 97-06 and the referenced EPRI SGMP guidelines provide recommendations concerning the inspection of tubes, which cover inspection equipment, baseline inspections, tube selection, sampling and frequency of inspection, methods of recording, required actions based on findings, and tube plugging. The minimum requirements for inservice inspection of steam generators, including plugging criteria, are established in Technical Specification 5.5.4.

The tube surveillance and degradation monitoring programs include provisions to maintain the compatibility of steam generator tubing with primary and secondary coolant to limit the steam generators' susceptibility to corrosion. These provisions are in accordance with NEI 97-06.

5.4.15 COMBINED LICENSE INFORMATION ITEMS

STD COL 5.4-1 This COL Item is addressed in Subsection 5.4.2.5.

5.4.16 REFERENCES

Insert the following information at the end of DCD Subsection 5.4.16.

201. Nuclear Energy Institute, “Steam Generator Program Guidelines,” NEI 97-06, Revision 2, May 2005.

CHAPTER 6

ENGINEERED SAFETY FEATURES

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CHAPTER 6
ENGINEERED SAFETY FEATURES

6.0 ENGINEERED SAFETY FEATURES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

6.1 ENGINEERED SAFETY FEATURES MATERIALS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6.1.1.2 Fabrication Requirements

Add the following information to the end of DCD Subsection 6.1.1.2:

STD COL 6.1-1

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 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 Subsection 5.2.3.

6.1.2.1.6 Quality Assurance Features

Replace the third paragraph under the subsection titled “Service Level I and Service Level III Coatings” within DCD Subsection 6.1.2.1.6 with the following information.

STD COL 6.1-2

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 (Reference 201) 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, 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 (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 (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.

Include a new second paragraph under the subsection titled “Service Level II Coatings” within DCD Subsection 6.1.2.1.6 with the following information.

Such 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.

Replace the second sentence of the third paragraph under the subsection titled “Service Level II Coatings” within DCD Subsection 6.1.2.1.6 with the following information.

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 (Reference 201), “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants,” and the guidance of ASTM D5163 (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.

6.1.3 COMBINED LICENSE INFORMATION ITEMS

6.1.3.1 Procedure Review

STD COL 6.1-1 This COL Item is addressed in Subsection 6.1.1.2.

6.1.3.2 Coating Program

STD COL 6.1-2 This COL Item is addressed in Subsection 6.1.2.1.6.

The following information supplements the information provided in DCD Subsection 6.1.4.

6.1.4 REFERENCES

201. ASTM D5144-08, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants.”

202. ASTM D5163-05a, “Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant.”

203. ASTM D7167-05, “Standard Guide for Establishing Procedures to Monitor the Performance of Safety-Related Coating Service Level III Lining Systems in an Operating Nuclear Power Plant.”

6.2 CONTAINMENT SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6.2.5.1 Design Basis

Add the following information at the end of DCD Subsection 6.2.5.1, as identified in Appendix A to NuStart Technical Report AP-TR-NS01-A, Rev 2, “Containment Leak Rate Test Program Description.”

STD COL 6.2-1 The Containment Leak Rate Test Program using 10 CFR Part 50, Appendix J Option B is established in accordance with NEI 94-01 (DCD Subsection 6.2.7, Reference 30), as modified and endorsed by the NRC in Regulatory Guide 1.163. Table 13.4-201 provides milestones for containment leak rate testing implementation.

6.2.5.2.2 System Operation

Add the following information at the end of the subsection "Scheduling and Reporting of Periodic Tests" within DCD Subsection 6.2.5.2.2, as identified in Appendix A to NuStart Technical Report AP-TR-NS01-A, Rev 2, “Containment Leak Rate Test Program Description.”

STD COL 6.2-1 Schedules for the performance of periodic Type A, B, and C leak rate tests are in accordance with NEI 94-01, as endorsed and modified by Regulatory Guide 1.163, and described below:

<u>Type A Tests</u>

A preoperational Type A test is conducted prior to initial fuel load. If initial fuel load is delayed longer than 36 months after completion of the preoperational Type A test, a second preoperational Type A test shall be performed prior to initial fuel load. The first periodic Type A test is performed within 48 months after the successful completion of the last preoperational Type A test. Periodic Type A tests are performed at a frequency of at least once per 48 months, until acceptable performance is established. The interval for testing begins at initial reactor operation. Each test interval begins upon completion of a Type A test and ends at the start of the next test. The extension of the Type A test interval is determined in accordance with NEI 94-01.

Type A testing is performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as successful completion of two consecutive Type A tests where the calculated performance leakage rate was less than 1.0 L_a. A preoperational Type A test may be used as one of the two Type A tests that must be successfully completed to extend the test interval, provided that an engineering analysis is performed to document why a preoperational Type A test can be treated as a periodic test. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

Type B Tests (Except Containment Airlocks)

Type B tests are performed prior to initial entry into Mode 4. Subsequent periodic Type B tests are performed at a frequency of at least once per 30 months, until acceptable performance is established. The test intervals for Type B penetrations may be increased based upon completion of two consecutive periodic as-found Type B tests where results of each test are within allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be 24 months or the nominal test interval (e.g., refueling cycle) for the component prior to implementing Option B of 10 CFR Part 50, Appendix J. An extended test interval for Type B tests may be increased to a specific value in a range of frequencies from greater than once per 30 months up to a maximum of once per 120 months. The extension of specific test intervals for Type B penetrations is determined in accordance with NEI 94-01.

Type B Tests (Containment Airlocks)

Containment airlock(s) are tested at an internal pressure of not less than P_{ac}. (Prior to a preoperational Type A test P_{ac}= P_a.) Subsequent periodic tests are performed at a frequency of at least once per 30 months. In addition, equalizing valves, door seals, and penetrations with resilient seals (i.e., shaft seals, electrical penetrations, view port seals and other similar penetrations) that are testable, are tested at a frequency of once per 30 months.

For periods of multiple containment entries where the airlock doors are routinely used for access more frequently than once every seven days (e.g., shift or daily inspection tours of the containment), door seals may be tested once per 30 days during this time period.

Airlock door seals are tested prior to a preoperational Type A test. When containment integrity is required, airlock door seals are tested within seven days after each containment access.

Type C Tests

Type C tests are performed prior to initial entry into Mode 4. Subsequent periodic Type C tests are performed at a frequency of at least once per 30 months, until adequate performance has been established. Test intervals for Type C valves may be increased based upon completion of two consecutive periodic as-found Type C tests where the result of each test is within allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive passing tests used to determine performance shall be 24 months or the nominal test interval (e.g., refueling cycle) for the valve prior to implementing Option B of 10 CFR Part 50, Appendix J. Intervals for Type C testing may be increased to a specific value in a range of frequencies from 30 months up to a maximum of 60 months. Test interval extensions for Type C valves are determined in accordance with NEI 94-01.

Reporting

A post-outage report is prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B and Type C tests, if performed during that outage. The report is available on-site for NRC review. The report shows that the applicable performance criteria are met, and serves as a record that continuing performance is acceptable.

Add the following subsection at the end of DCD Subsection 6.2.5.2.2, as identified in Appendix A to NuStart Technical Report AP-TR-NS01-A, Rev 2, “Containment Leak Rate Test Program Description.”

STD COL 6.2-1 Acceptance Criteria

Acceptance criteria for Type A, B and C Tests are established in Technical Specification 5.5.8.

6.2.6 COMBINED LICENSE INFORMATION FOR CONTAINMENT LEAK RATE TESTING

STD COL 6.2-1 This COL item is addressed in Subsections 6.2.5.1 and 6.2.5.2.2.

6.3 PASSIVE CORE COOLING SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6.3.8 COMBINED LICENSE INFORMATION

6.3.8.1 Containment Cleanliness Program

Insert the following information at the end of DCD Subsection 6.3.8.1:

This COL Item is addressed below.

STD COL 6.3-1 Administrative procedures implement the containment cleanliness program. Implementation of the program minimizes the amount of debris left in containment following personnel entry and exits. The program is consistent with the containment cleanliness program limits discussed in DCD Subsection 6.3.8.1. The program includes, as a minimum, the following:

Responsibilities

The program defines the organizational responsibilities for implementing the program; defines personnel and material controls; and defines the inspection and reporting requirements.

Implementation

Containment Entry/Exit

- Controls to account for the quantities and types of materials introduced into the containment.
- Limits on the types and quantities of materials, including scaffolding and tools, to ensure adequate accountability controls. This may be accomplished by the work management process. Storage of aluminum is prohibited without engineering authorization. Cardboard boxes or miscellaneous packing material is not brought into containment without approval.
- If entries are made at power, prohibited materials and limits on quantities of materials that may generate hydrogen are established.
- Controls for loose items, such as keys and pens, which could be inadvertently left in containment.

- Methods and controls for securing any items and materials left unattended in containment.
- Administrative controls for accounting for tools, equipment and other material are established.
- Administrative controls for accounting of the permanent removal of materials previously introduced into the containment.
- Limits on the types and quantities of materials, including scaffolding and tools, that may be left unattended in containment during outages and power operation. Types of materials considered are tape, labels, plastic film, and paper and cloth products.
- Requirements and actions to be taken for unaccounted for material.
- Requirements for final containment cleanliness inspections consistent with the design bases provided in DCD Subsection 6.3.8.1.
- Record keeping requirements for entry/exit logs.

Housekeeping

Housekeeping procedures require that work areas be maintained in a clean and orderly fashion during work activities and returned to original conditions (or better) upon completion of work.

Sampling Program

A sampling program is implemented consistent with NEI Guidance Report 04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology” as supplemented by the NRC in the “Safety Evaluation by The Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Nuclear Energy Institute Guidance Report (Proposed Document Number NEI 04-07), ‘Pressurized Water Reactor Sump Performance Evaluation Methodology.’” Latent debris sampling is implemented before startup. The sampling is conducted after containment exit cleanliness inspections to provide reasonable assurance that the plant latent debris design bases are met. Sampling frequency and scope may be adjusted based on sampling results. Results are evaluated post-start up and any nonconforming results will be addressed in the Corrective Action Program.

6.4 HABITABILITY SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6.4.3 SYSTEM OPERATION

Add the following information at the end of DCD Subsection 6.4.3:

STD COL 6.4-2 Generic Issue 83 addresses the importance of maintaining control room habitability following an accidental release of external toxic or radioactive material or smoke and the capability of the control room operators to safely control the reactor. Procedures and training for control room habitability are written in accordance with Section 13.5 for control room operating procedures, and Section 13.2 for operator training. The procedures and training are verified to be consistent to the intent of Generic Issue 83.

The procedures and training address the toxic chemical events addressed in Sections 2.2 and 6.4 consistent with the guidance provided in regulatory position C.5 of Regulatory Guide 1.78, including arrangements with Federal, State, and local agencies or other cognizant organizations for the prompt notification of the nuclear power plant when accidents involving hazardous chemicals occur within five miles of the plant. The procedures include the conduct of periodic surveys of stationary and mobile sources of hazardous chemicals affecting the evaluations consistent with the guidance provided in regulatory position 2.5 of Regulatory Guide 1.196. The procedures include appropriate reviews of the configuration of the control room envelope and habitability systems consistent with the guidance provided in regulatory position 2.2.1 of Regulatory Guide 1.196. The procedures also include periodic assessments of the control room habitability systems' material condition, configuration controls, safety analyses, and operating and maintenance procedures consistent with the guidance provided in regulatory position 2.2.1 of Regulatory Guide 1.196.

Procedures for testing and maintenance are consistent with the design requirements of the DCD including the guidance provided in regulatory position 2.7.1 of Regulatory Guide 1.196.

6.4.4 SYSTEM SAFETY EVALUATION

Insert the following information at the end of the eighth paragraph of DCD Subsection 6.4.4.

VEGP COL 6.4-1 STD COL 6.4-1

Table 6.4-201 provides additional details regarding the evaluated onsite chemicals.

Insert the following subsections at the end of DCD Subsection 6.4.4.

6.4.4.1 Dual Unit Analysis

STD SUP 6.4-1

Credible events that could put the control room operators at risk from a dose standpoint at a single AP1000 unit have been evaluated and addressed in the DCD. The dose to the control room operators at an adjacent AP1000 unit due to a radiological release from another unit is bounded by the dose to control room operators on the affected unit. While it is possible that a unit may be downwind in an unfavorable location, the dose at the downwind unit would be bounded by what has already been evaluated for a single unit AP1000. Simultaneous accidents at multiple units at a common site are not considered to be a credible event.

VEGP SUP 6.4-2

The hazard due to the effects of a Design Basis Accident (DBA) from Units 1 and 2 is discussed in ESPA SSAR Subsection 2.2.3.4.

6.4.4.2 Toxic Chemical Habitability Analysis

VEGP COL 6.4-1 VEGP COL 9.4-1b

Offsite chemicals are evaluated in ESPA SSAR Subsection 2.2.3. Site-specific onsite chemicals are evaluated in Subsection 2.2.3 and ESPA SSAR Subsection 2.2.3. Evaluation results show that there are no toxic hazards to Units 3 and 4 control room personnel.

During a toxic gas emergency, the control room operators have the option of manually actuating the emergency habitability system (DCD Subsection 1.9.4.2.3, Issue 83). This action activates isolation dampers for the control room and supplies positive internal pressure and breathing air via bottled gas. Because the emergency habitability system isolates and pressurizes the control room, the activation of the system stabilizes and begins to decrease the concentration of toxic gas in the control room at the actuation value.

Normal HVAC operation would resume after the gas cloud passes the site, rapidly dropping the remaining elevated levels of toxic gas in the control room to non-detectable levels.

6.4.7 COMBINED LICENSE INFORMATION

VEGP COL 6.4-1
STD COL 6.4-1

This COL Item is addressed in Subsections 2.2.3.2.3.1, 2.2.3.2.3.2, 2.2.3.3, 6.4.4, and 6.4.4.2.

STD COL 6.4-2

This COL Item is addressed in Subsection 6.4.3.

Table 6.4-201 (Sheet 1 of 4)
Main Control Room Habitability Evaluations of Onsite Toxic Chemicals(1)

STD COL 6.4-1

A — Standard Onsite Toxic Chemicals

Evaluated Material	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR Intake	Evaluated Location	MCR Habitability Impact Evaluation
Hydrogen	Gas	500 scf	126.3 ft	Yard at turbine building	MCR
Hydrogen	Liquid	1500 gal	577 ft	Gas storage	MCR
Nitrogen	Liquid	3000 gal	577 ft	Gas storage	MCR
Carbon Dioxide (CO_2)	Liquid	6 tons	577 ft	Gas storage	MCR
Oxygen Scavenger [Hydrazine]	Liquid	1600 gal	203 ft	Turbine building	IH
pH Addition [Morpholine]	Liquid	1600 gal	203 ft	Turbine building	IH
Sulfuric Acid	Liquid	800 gal	203 ft	Turbine building	IH
Sulfuric Acid	Liquid	20,000 gal	436 ft	CWS area	IH
Sodium Hydroxide	Liquid	800 gal	203 ft	Turbine building	S
Sodium Hydroxide	Liquid	20,000 gal	436 ft	CWS area	S
Fuel Oil	Liquid	60,000 gal	197 ft	DG fuel oil storage tank, DG building, Annex building	IH

Table 6.4-201 (Sheet 2 of 4)

Main Control Room Habitability Evaluations of Onsite Toxic Chemicals[1]

STD COL 6.4-1

A — Standard Onsite Toxic Chemicals

Evaluated Material	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR Intake	Evaluated Location	MCR Habitability Impact Evaluation
Corrosion Inhibitor [Sodium Molybdate]	Liquid	800 gal	203 ft	Turbine building	S
Corrosion Inhibitor [Sodium Molybdate]	Liquid	10,000 gal	436 ft	CWS area	S
Scale Inhibitor [Sodium Hexametaphosphate]	Liquid	800 gal	203 ft	Turbine building	S
Scale Inhibitor [Sodium Hexametaphosphate]	Liquid	10,000 gal	436 ft	CWS area	S
Biocide/Disinfectant [Sodium hypochlorite]	Liquid	800 gal	203 ft	Turbine building	S
Biocide/Disinfectant [Sodium hypochlorite]	Liquid	10,000 gal	436 ft	CWS area	S
Algaecide [Ammonium comp. polyethoxylate]	Liquid	800 gal	203 ft	Turbine building	S
Algaecide [Ammonium comp. polyethoxylate]	Liquid	10,000 gal	436 ft	CWS area	S

Table 6.4-201 (Sheet 3 of 4)
Main Control Room Habitability Evaluations of Onsite Toxic Chemicals(1)

VEGP COL 6.4-1

B — Site Specific Onsite Toxic Chemicals

Evaluated Material	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR Intake	Evaluated Location	MCR Habitability Impact Evaluation
pH Control [Methoxypropylamine (MPA)]	Liquid	800 gal	211 ft	Turbine building	IH, MCR
Silt dispersant [Proprietary tagged high strength polymer]	Liquid	800 gal	211 ft	Turbine building	S
Silt dispersant [Proprietary tagged high strength polymer]	Liquid	10,000 gal	802 ft	CWS area	S
Corrosion inhibitor [Proprietary blend of Phosphonate, Phosphinosuccinic Oligomer (PSO), and Phosphoric Acid]	Liquid	800 gal	211 ft	Turbine building	S
Corrosion inhibitor [Proprietary blend of Phosphonate, Phosphinosuccinic Oligomer (PSO), and Phosphoric Acid]	Liquid	10,000 gal	802 ft	CWS area	S
Biocide [Stabilized Bromine)	Liquid	800 gal	211 ft	Turbine building	S
Biocide [Stabilized Bromine)	Liquid	10,000 gal	802 ft	CWS area	S

Table 6.4-201 (Sheet 4 of 4)
Main Control Room Habitability Evaluations of Onsite Toxic Chemicals(1)

VEGP COL 6.4-1

B — Site Specific Onsite Toxic Chemicals

Evaluated Material	Evaluated State	Evaluated Maximum Quantity	Evaluated Minimum Distance to MCR Intake	Evaluated Location	MCR Habitability Impact Evaluation
Biocide [Sodium Bromide)	Liquid	800 gal	211 ft	Turbine building	S
Biocide [Sodium Bromide)	Liquid	10,000 gal	802 ft	CWS area	S
Detoxification agent [Ammonium Bisulfite]	Liquid	1000 gal	211 ft	Turbine building	IH, MCR

Notes:

STD COL 6.4-1

1) This table supplements DCD Table 6.4-1. Quantities are by largest evaluated container content for the evaluated location per unit. Quantities and distances are bounding evaluation values and may not be actual amounts and distances. Smaller quantities of a chemical at further distances from the MCR air intake are not shown on this table. Actual site locations are confirmed to be at or beyond the evaluated distance.

S - Chemicals with an Impact Evaluation designation of “S” for the MCR Habitability Impact Evaluation were evaluated and screened out based on the chemical properties, distance, and quantities.

IH - Chemicals with an Impact Evaluation designation of “IH” indicates the evaluation of this chemical considered the design detail of the main control room intake height.

MCR - Chemicals with an Impact Evaluation designation of “MCR” indicates the evaluation of this chemical considered design details of the main control room such as volume, envelope boundaries, ventilation systems, and occupancy factor.

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

6.6 INSERVICE INSPECTION OF CLASS 2, 3, AND MC COMPONENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following to DCD Section 6.6 ahead of Subsection 6.6.1 heading:

STD COL 6.6-1 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 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 ASME Code cases listed in 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)).

6.6.1 COMPONENTS SUBJECT TO EXAMINATION

Add the following to the end of DCD Subsection 6.6.1:

STD COL 6.6-1 Class 2 and 3 components are included in the equipment designation list and the line designation list contained in the inservice inspection program.

6.6.2 ACCESSIBILITY

Revise the first and last sentences of the third paragraph in DCD Subsection 6.6.2 to add supplemental information as follows:

STD SUP 6.6-1 Considerable experience has been drawn on in designing, locating, and supporting Quality Group B and C (ASME Class 2 and 3) and Class MC pressure-retaining components to permit pre-service and inservice inspection required by Section XI of the ASME Code. Factors such as examination requirements, examination techniques, accessibility, component geometry, and material selections are used in establishing the designs. The inspection design goals are 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. There are no Quality Group B and C components or Class MC components, which require inservice inspection during reactor operation.

Add the following to the end of DCD Subsection 6.6.2:

STD COL 6.6-2 During the construction phase of the project, anomalies and construction issues are addressed using change control procedures. Modifications reviewed following design certification adhere to the same level of review as the certified design per 10 CFR Part 50, Appendix B as implemented by the Westinghouse Quality Management System (QMS). The QMS requires that changes to approved design documents, including field changes, are subject to the same review and approval process as the original design. This explicitly requires the field change process to follow the same level of review that was required during the design process. Accessibility and inspectability are key components of the design process.

Control of accessibility for inspectability and testing during post-design certification activities is provided via procedures for design control and plant modifications.

6.6.3 EXAMINATION TECHNIQUES AND PROCEDURES

Add the following Subsections 6.6.3.1, 6.6.3.2 and 6.6.3.3 to the end of DCD Subsection 6.6.3:

6.6.3.1 Examination Methods

Visual Examination

STD COL 6.6-1 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 meet the requirements of IWA-5240.

Where direct visual VT-1 examinations are conducted without the use of mirrors or with other viewing aids, clearance is provided in accordance with Table IWA-2210-1.

Surface Examination

Magnetic particle, liquid penetrant, and eddy current examination techniques are performed in accordance with ASME Section XI, IWA-2221, IWA-2222, and

IWA-2223 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 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.

Ultrasonic Examination

Volumetric ultrasonic direct examination is performed in accordance with ASME Section XI, IWA-2232, which references mandatory Appendix I.

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 method. This provision allows for the use of newly developed examination 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.

6.6.3.2 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.

6.6.3.3 Relief Requests

The specific areas where the applicable ASME Code requirements cannot be met are identified after the examinations are performed. Should relief requests be required, they will be developed through the regulatory process and submitted to the NRC for approval in accordance with 10 CFR 50.55a(a)(3) or 50.55a(g)(5). The relief requests include appropriate justifications and proposed alternative inspection methods.

6.6.4 INSPECTION INTERVALS

Add the following to the end of DCD Subsection 6.6.4:

STD COL 6.6-1 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. Period one comprises the first three years of the interval, period two comprises the next four years of the interval, and period three comprises the remaining three years of the inspection interval. The periods within each inspection interval may be extended by as much as one year to permit inspections to be concurrent with plant outages. The adjustment of period end dates shall not alter the rules and requirements for scheduling inspection intervals. It is intended that inservice examinations be performed during normal plant outages, such as refueling shutdown or maintenance shutdowns occurring during the inspection interval.

6.6.6 EVALUATION OF EXAMINATION RESULTS

Add the following new paragraph at the end of DCD Subsection 6.6.6:

STD COL 6.6-1 Components containing flaws or relevant conditions and accepted for continued service in accordance with the requirements of IWC-3122.3 or IWC-3132.3 for Class 2 components, IWD-3000 for Class 3 components, IWE-3122.3 for Class MC components, or IWF-3112.2 or IWF-3122.2 for component supports, are subjected to successive period examinations in accordance with the requirements of IWC-2420, IWD-2420, IWE-2420, or IWF-2420, respectively. Examinations that reveal flaws or relevant conditions exceeding Table IWC-3410-1, IWD-3000, IWE-3000, or IWF-3400 acceptance standards are extended to include additional examinations in accordance with the requirements of IWC-2430, IWD-2430, or IWF-2430, respectively.

6.6.9 COMBINED LICENSE INFORMATION ITEMS

6.6.9.1 Inspection Programs

STD COL 6.6-1 This COL Item is addressed in Section 6.6 introduction, and in Subsections 6.6.1, 6.6.3.1, 6.6.3.2, 6.6.3.3, 6.6.4, and 6.6.6.

6.6.9.2 Construction Activities

STD COL 6.6-2 This COL Item is addressed in Subsection 6.6.2.

APPENDIX 6A
FISSION PRODUCT DISTRIBUTION IN THE AP1000 POST-DESIGN BASIS ACCIDENT CONTAINMENT ATMOSPHERE

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 7

INSTRUMENTATION AND CONTROLS

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CHAPTER 7
INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

7.1.6.1 Setpoint Calculations for Protective Functions

STD COL 7.1-1 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.

7.2 REACTOR TRIP

This section of the referenced DCD is incorporated by reference with no departures or supplements.

7.3 ENGINEERED SAFETY FEATURES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

This section of the referenced DCD is incorporated by reference with no departures or supplements.

7.5 SAFETY-RELATED DISPLAY INFORMATION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

7.5.2 VARIABLE CLASSIFICATIONS AND REQUIREMENTS

Add the following paragraph at the end of DCD Subsection 7.5.2.

STD COL 7.5-1 FSAR Table 7.5-201 supplements DCD Table 7.5-1 and provides variable data shown in the DCD Table as "site specific."

7.5.3.5 Type E Variables

Add the following paragraphs at the end of DCD Subsection 7.5.3.5.

STD COL 7.5-1 FSAR Table 7.5-202 supplements DCD Table 7.5-8 and provides variable data shown in the DCD Table as "site specific."

7.5.5 COMBINED LICENSE INFORMATION

STD COL 7.5-1
VEGP COL 7.5-1 This COL item is addressed in Subsection 7.5.2 and Table 7.5-201, and in Subsection 7.5.3.5 and Table 7.5-202.

VEGP COL 7.5-1

TABLE 7.5-201
POST-ACCIDENT MONITORING SYSTEM[a]

Variable	Range/Status	Type/ Category	Qualification		Number of Instruments Required	Power Supply	QDPS Indication	Remarks
			Environmental	Seismic				
Boundary environs radiation • Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability) • Radiation (portable instrumentation) • Radioactivity (portable instrumentation)	10^{-9} to 10^{-3} μCi/cc 10^{-3} to 10^{4} R/hr, photons 10^{-3} to 10^{4} rads/hr, beta and low-energy photons Multichannel gamma ray spectrometer	C3, E3	None	None	N/A	Non-1E	No	Conforms to RG 1.97, Revision 3
Meteorological parameters • Wind Speed • Wind Direction • Differential Temperature	0 – 100 mph (±0.5 mph) 0° – 540° (±2.43°) -9.4°F to 19.4°F (±0.212°F)	E3	None	None	2 (1@ 10 m and 1 @ 60 m) 2 (1@ 10 m and 1 @ 60 m) 1 (10 – 60 m)	Non-1E	No	Conforms to RG 1.97, Revision 3

(a) This Table supplements DCD Table 7.5-1 and provides the site specific information in the remarks column of DCD Table 7.5-1.

VEGP COL 7.5-1

TABLE 7.5-202
SUMMARY OF TYPE E VARIABLES[(a)]

Function Monitored	**Variable**	**Type/ Category**
Environs Radiation and Radioactivity	Plant Environs radiation levels and airborne radioactivity	E3
Meteorology	Wind speed, wind direction, and estimation of atmospheric stability (based on vertical temperature difference)	E3

(a) This Table supplements DCD Table 7.5-8 and provides the site specific information noted in the variable column of DCD Table 7.5-8.

7.6 INTERLOCK SYSTEMS IMPORTANT TO SAFETY

This section of the referenced DCD is incorporated by reference with no departures or supplements.

7.7 CONTROL AND INSTRUMENTATION SYSTEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 8

ELECTRIC POWER

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CHAPTER 8
ELECTRIC POWER

8.1 INTRODUCTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

8.1.1 UTILITY GRID DESCRIPTION

Replace the existing information in DCD Subsection 8.1.1 with the following information.

VEGP SUP 8.1-1 VEGP is interconnected to the Southern Balancing Authority Area (SBAA) transmission grid operated by Southern Company Transmission (SCT). The SBAA transmission grid consists of transmission facilities owned by Alabama Power Company, Georgia Power Company, Gulf Power Company, and Mississippi Power Company, all subsidiaries of the Southern Company and Georgia Transmission Corporation, the Municipal Electric Authority of Georgia and Dalton Utilities, who along with Georgia Power Company are participants in and make up the Georgia Integrated Transmission System. The SBAA transmission grid is interconnected with Duke Power Company, South Carolina Electric and Gas Company, South Carolina Public Service Authority (Santee Cooper), Florida Power and Light Company, Progress Energy, City of Tallahassee, Entergy Gulf States, Entergy Louisiana, Mississippi Power and Light, and the Tennessee Valley Authority. The SBAA transmission grid interconnects hydro plants, fossil-fueled plants, and nuclear plants supplying electric energy over a transmission grid consisting of various voltages up to 500 kV.

VEGP Units 3 and 4 are located in eastern Burke County, Georgia, approximately 26 miles southeast of Augusta, Georgia and 100 miles northwest of Savannah, Georgia, directly across the Savannah River from the US Department of Energy's Savannah River Site in Barnwell County, South Carolina. VEGP Units 1 and 2, which are two Westinghouse Electric Company, LLC (Westinghouse) pressurized water reactors (PWRs), have been in commercial operation since 1987 and 1989, respectively. Plant Wilson, a six-unit oil-fueled combustion turbine facility owned by Georgia Power Company (GPC), is also located on the VEGP site. VEGP Units 3 and 4 are adjacent to and west of VEGP Units 1 and 2. VEGP Units 1 and 2 are co-owned by Georgia Power Company, Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia, and the City of Dalton, Georgia. Southern Nuclear Operating Company (SNC) is the licensed operator of the nuclear facilities at the VEGP site, with control of the nuclear facilities, including complete authority to regulate any and all access and activity within the plant exclusion area boundary. GPC and SNC are subsidiaries of Southern Company.

VEGP Unit 3 is connected to the Units 1, 2 and 3, 230/500 kV switchyard at the 230 kV level. The 230 kV and 500 kV levels of the Units 1, 2 and 3, 230/500 kV switchyard are arranged in a breaker-and-a-half configuration and are interconnected through two, 230/500 kV autotransformers. VEGP Unit 4 is connected to the Unit 4, 500 kV switchyard. This switchyard is also arranged in a breaker-and-a-half configuration. The Unit 4, 500 kV switchyard is connected to the 500 kV section of the Units 1, 2 and 3, 230/500 kV switchyard by overhead lines. The Reserve Auxiliary Transformers (RATs) for Units 3 and 4 are supplied by two overhead lines from a 230 kV switchyard with a ring bus configuration. A portion of Unit 3 RAT “A” supply line is underground between Unit 4 and Unit 3. Five, 230 kV and three, 500 kV transmission lines connect the VEGP high voltage switchyards to the remainder of the SBAA transmission grid.

8.1.4.3 Design Criteria, Regulatory Guides, and IEEE Standards

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VEGP SUP 8.1-2 Add the following information between the second and third paragraphs of this subsection.

Offsite and onsite ac power systems' conformance to Regulatory Guides and IEEE Standards identified by DCD Table 8.1-1 as site-specific and to other applicable Regulatory Guides is as indicated in Table 8.1-201.

VEGP SUP 8.1-2

Table 8.1-201
Site-Specific Guidelines For Electric Power Systems

Criteria				Applicability (FSAR[a] Section/Subsection) 8.2	8.3.1	8.3.2	Remarks
1.	Regulatory Guides						
	a.	RG 1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants			G	Battery Service tests are performed in accordance with the Regulatory Guide.
	b.	RG 1.155	Station Blackout				Not applicable[b]
	c.	RG 1.204	Guidelines for Lightning Protection of Nuclear Power Plants	G	G		Implemented via IEEE 665.
	d.	RG 1.206	Combined License Applications for Nuclear Power Plants (LWR Edition)	G	G	G	
2.	Branch Technical Positions						
	a.	BTP 8-3 (BTP ICSB-11 in DCD)	Stability of Offsite Power Systems	G			Stability Analysis of the Offsite Power System is performed in accordance with the BTP.

a) "G" denotes guidelines as defined in NUREG-0800, Rev. 3, Table 8-1 (SRP). No letter denotes "Not Applicable."
b) Station Blackout and the associated guidelines were addressed as a design issue in the DCD.

8.2 OFFSITE POWER SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

8.2.1 SYSTEM DESCRIPTION

Delete the first, second, and sixth paragraphs and the first and last sentences of the fourth paragraph, of DCD Section 8.2.1. Add the following information before the fifth paragraph of DCD Subsection 8.2.1.

VEGP COL 8.2-1 The Southern Balancing Authority Area (SBAA) transmission grid supplies the off-site AC power (preferred and maintenance power) for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 via three, high voltage switchyards located north of the facility. The three, high voltage switchyards are defined as follows:

- Units 1, 2 and 3, 230/500 kV switchyard
- Unit 4, 500 kV switchyard
- Units 3 and 4, Reserve Auxiliary Transformer (RAT) supply, 230 kV switchyard

The interconnection of the three switchyards, and the 230 kV and 500 kV transmission lines are shown on Figures 8.2-201 and 8.2-202.

VEGP Units 3 and 4 are connected into a transmission grid supplying large load centers. Unit 3 is tied into the 230 kV transmission grid via the Units 1, 2 and 3, 230/500 kV switchyard. Unit 4 is tied into the 500 kV transmission grid via the Unit 4, 500 kV switchyard.

The Units 3 and 4, RAT supply, 230 kV switchyard, consists of 4 breakers installed in a ring bus configuration. The Augusta Newsprint transmission line and the Units 1, 2 and 3, 230/500 kV switchyard are connected to the Units 3 and 4, RAT supply, 230kV switchyard.

230 kV Overhead Transmission Lines

Five, 230 kV overhead transmission lines connect the Units 1, 2 and 3, 230/500 kV switchyard to other substations throughout the transmission grid. Each 230 kV transmission line is connected to a Georgia Integrated Transmission System or a South Carolina Electric and Gas (SCEG) substation. The five, 230 kV transmission lines originate at the VEGP switchyards and connect to various substations as shown below.

230 kV Line	Termination Point	Length (miles)	Thermal Rating (MVA)
Augusta Newsprint	Augusta Newsprint Sub	20	596
Goshen White	Goshen Sub	19	866
Goshen Black	Goshen Sub	19	866
SCEG	Savannah River Plant Sub	22	1020
Plant Wilson	Plant Wilson Sub	1	718

The Augusta Newsprint, Goshen White and Goshen Black 230 kV lines are on a common right-of-way for the first 18.75 miles from the plant. The lines exit the north side of the switchyard and leave the plant site to the northwest. Approximately 0.5 miles from the plant, the Units 3 and 4, RAT supply, 230 kV switchyard is installed along the Augusta Newsprint transmission line.

The SCEG 230 kV line exits the north side of the switchyard, crosses the Savannah River and continues to the Savannah River Plant Substation in South Carolina.

The 230 kV transmission line from the Unit 3 generator to the Units 1, 2 and 3, 230/500 kV switchyard travels under these four transmission lines. All high voltage equipment and conductors are designed to meet the requirements of the National Electrical Safety Code (NESC) and Georgia Power Company (GPC) engineering standards. Electrical clearances phase-to-phase and phase-to-ground are determined by NESC and engineering requirements.

The Plant Wilson 230 kV line exits the north side of the switchyard and leaves the plant site traveling southeast to the Wilson switchyard less than 1 mile away.

<u>500 kV Overhead Transmission Lines</u>

Three 500 kV overhead transmission lines connect the Units 1, 2 and 3, 230/500 kV switchyard and the Unit 4, 500 kV switchyard to other substations throughout the SBAA transmission grid. Each 500 kV transmission line is connected to a Georgia Integrated Transmission System substation. The three, 500 kV transmission lines originate at the VEGP switchyards and connect to various substations as shown below.

500kV Line	Termination Point	Length (miles)	Thermal Rating (MVA)
West McIntosh	West McIntosh Sub	69	2439
Scherer	Scherer Sub	154	2701
Thomson	Thomson Sub	52	2701

The Thomson 500 kV line exits the switchyard to the north and crosses over the 230 kV transmission line from the Unit 3 generator and then west over the 230 kV transmission lines to the Units 3 and 4 RATs. All high voltage equipment and conductors are designed to meet the requirements of the National Electrical Safety Code (NESC) and GPC engineering standards. After leaving the plant, the Thomson and Scherer 500 kV lines are on a common right-of-way for approximately the first 5 miles. The lines exit the north side of the switchyard and leave the plant site to the west.

The West McIntosh 500 kV line exits the north side of the switchyard and leaves the plant site traveling to the west before changing direction to the southeast.

All transmission lines connected to the VEGP switchyards are suspended from steel, lattice-type towers designed to provide clearances consistent with NESC and GPC engineering standards. Electrical clearances phase-to-ground are determined by NESC and engineering requirements, but are not less than 27 feet for 230 kV transmission lines and 32 feet for 500 kV transmission lines. Transmission towers include overhead ground wires. All towers are grounded with either ground rods or a counterpoise system.

VEGP CDI

A transformer area containing the generator step-up transformers (GSU), the unit auxiliary transformers (UATs), and reserve auxiliary transformers (RATs) is located next to each turbine building.

8.2.1.1 Transmission Switchyard

Replace the information in DCD Subsection 8.2.1.1 with the following information.

VEGP COL 8.2-1

A 230 kV air insulated switchyard, installed along the Augusta Newsprint transmission line, supplies power to the Units 3 and 4 RATs and is located approximately 3500 feet to the north of Units 3 and 4. This arrangement provides the power supply to the RATs from the Units 1, 2 and 3, 230/500 kV switchyard or directly from the Augusta Newsprint transmission line. Two overhead transmission lines connect the Units 3 and 4, RAT supply, 230 kV switchyard to the two RATs for Unit 3 and the two RATs for Unit 4. One overhead transmission line supplies RAT “A” for Units 3 and 4 and the other overhead transmission line supplies RAT “B” for Units 3 and 4. A portion of Unit 3 RAT “A” supply line is underground between Unit 4 and Unit 3. The RATs may be used to distribute power for plant auxiliaries when the GSUs or UATs are out of service. The Units 1, 2 and 3, 230/500 kV switchyard is used to transmit electrical power output from Unit 3 to the SBAA transmission grid. The 230 kV section of the switchyard is interconnected to the 500 kV section through two, 230/500 kV autotransformers.

A 500 kV air insulated switchyard is located 500 feet north of Unit 4 and 1500 feet to the west of the Units 1, 2 and 3, 230/500 kV switchyard. The Unit 4, 500kV switchyard is interconnected to the 500 kV section of the Units 1, 2 and 3, 230/500 kV switchyard by two overhead lines. These two lines cross over the 230 kV lines for the Unit 3 generator and the supply line for the Units 3 and 4 RAT “B”. The Unit 4, 500 kV switchyard is used to transmit electrical power output from Unit 4 to the SBAA transmission grid.

The Unit 3 generator is connected to the Units 1, 2 and 3, 230/500 kV switchyard by overhead tie lines that exit the north side of the switchyard and travel west, and then south to the GSU. The Unit 4 generator is connected to the Unit 4, 500kV switchyard by overhead tie lines that exit the south side of the switchyard and travel south to the GSU.

The high voltage circuit breakers in all three switchyards are sized with sufficient continuous current carrying capacity and fault interrupting capability to perform their intended function. The switchyard disconnect switches are rated equal to or greater than the continuous current basis of their associated circuit breakers. The Units 1, 2 and 3, 230/500 kV switchyard, the Unit 4, 500 kV switchyard and the Units 3 and 4, RAT supply, 230 kV switchyard are shown on the offsite power system one-line diagram and the switchyard general arrangement, Figures 8.2-201 and 8.2-202 respectively.

Failure Analysis

VEGP SUP 8.2-1 The design of the offsite power system provides for a robust system that supports reliable power production. Offsite power is not required to meet any safety function, and physical independence is not necessary. The certified design has been granted a partial exemption to GDC 17 by the NRC. Multiple, reliable transmission circuits are provided to support operation of the facility. Neither the accident analysis nor the Probabilistic Risk Assessment has identified the non-safety related offsite power system as risk significant for normal plant operation.

VEGP Units 3 and 4 are supplied with off-site power from the SBAA 230 kV and 500 kV transmission grid via two separate switchyard buses and backfed through the GSUs. The VEGP switchyards are connected to eight transmission lines. No single transmission line is designated as the preferred circuit, but analysis shows that with any one of these transmission lines out of service, the transmission grid can supply the switchyard with sufficient power for the safety related systems and other auxiliary loads for normal, abnormal and accident conditions on Units 3 and 4.

For the portions of the switchyards associated with VEGP Units 3 and 4, a failure modes and effects analysis (FMEA) confirmed that a single initiating event, such as a transmission line fault, plus a single breaker not operating, does not cause failure of more than one single transmission line, or a loss of offsite power to either Unit 3 or 4 via the GSU. This evaluation recognizes that a single failure of some switchyard components could directly cause the loss of the switchyard feed to the GSU, such as a fault on this feed. Evaluated events include 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. In summary:

- In the event of a fault on a 230 kV or 500 kV transmission line, the associated bus breakers trip and both buses stay energized and both units continue operation.
- In the event of a fault on a 230 kV or 500 kV transmission line concurrent with a stuck bus breaker, the affected bus differential relays cause circuit breakers on the affected bus to trip and thereby isolate the affected bus. Both units continue operation through the non-affected bus.
- In the event of a 230 kV or 500 kV bus fault, bus differential relays sense the fault and the breakers associated with the affected bus trip, thereby isolating the faulted bus. In this event, both units continue normal operation through the non-affected bus.
- In the event of a 230 kV or 500 kV bus fault concurrent with a stuck breaker, the adjacent breaker senses the fault and trips, which isolates the faulted bus. If the stuck breaker is associated with the output of either the Unit 3 or Unit 4 GSU, opening of the adjacent breaker interrupts power to the associated GSU and UATs resulting in a reactor trip. The unaffected unit continues operation through the non-affected bus.
- In the event of transmission line relay mis-operation, both associated switchyard breakers trip, isolating one of the transmission lines from the yard. In this event, both switchyard buses stay energized and both units continue normal operation.
- In the case of a loss of DC control power, the loss of control power to a breaker or to a transmission line primary relay is compensated for by

redundant trip coils powered from a different source which allows the protective function to occur. Both units continue normal operation.

- In the event of a fault on the 230 kV Augusta Newsprint line concurrent with a stuck breaker, power will be lost to either the Units 3 and 4 RAT “A” or Units 3 and 4 RAT “B” by tripping the adjacent circuit breaker with the stuck breaker relay protection schemes. In this event, both units continue normal operation and the maintenance power supply to the “A” or “B” RATs will be lost.

The results of the analysis show that a single fault in any section of the 230 kV or 500 kV bus is cleared by the adjacent breakers and does not interrupt operation of the remaining part of the switchyard bus or the connection of the unaffected transmission lines. A bus fault with a stuck breaker associated with the output of either the Unit 3 or Unit 4 GSU causes the loss of power to, and a reactor trip in the associated unit. A bus fault concurrent with any other stuck breaker does not cause a loss of power to either unit.

An analysis was performed of transmission line crossings within the area of the Vogtle site. Sixteen line crossings were evaluated to demonstrate that offsite power would be available to both Unit 3 and Unit 4 from at least one of the three available offsite power supplies to each unit. A nonmechanistic failure was assumed for each of the 16 transmission lines (a line is considered to be any one of the three phases) allowing it to fall on the line or lines immediately below it. In three cases, the falling line was assumed to contact two lines below. In all, 13 separate cases of falling transmission lines were evaluated. No single failures of protective relaying or breakers were assumed in this evaluation. The evaluation demonstrated that, in each case, at least one offsite power supply remained available to both Unit 3 and Unit 4.

Transmission System Operator (TSO)

Southern Company Transmission (SCT) is the TSO within the SBAA and is responsible for the safe and reliable operation of the SBAA transmission grid. The SBAA is located within the SERC Reliability Corporation, one of the regional corporations within the North American Electric Reliability Corporation (NERC). SCT has responsibility for Transmission Planning and Operation of the bulk power transmission system. The Operation is performed by the Georgia Transmission Control Center (GCC) in Atlanta, Georgia and Bulk Power Operations (BPO) organization. The BPO control center is also known as the Power Coordination Center (PCC) and is located in Birmingham, Alabama.

SCT and VEGP have an agreement and protocols in place to provide safe and reliable operation of the transmission grid and equipment at VEGP Units 3 and 4. Elements of this agreement are implemented in accordance with the procedures of both parties.

The TSO establishes a voltage schedule for the 230 kV and 500 kV switchyards. VEGP Units 3 and 4, while generating, are expected to supply or absorb reactive power to help regulate voltage in the 230 kV and 500 kV switchyards in accordance with TSO voltage schedule criteria. VEGP maintains switchyard voltage such that steady state voltage on the 26 kV generator terminals is within 0.95 – 1.05 per unit (p.u.) of its nominal value.

VEGP SUP 8.2-1 VEGP provides the TSO with a nuclear plant interface agreement that specifies the detailed voltage and other requirements necessary to ensure safe and reliable operation of VEGP. The minimum and maximum switchyard voltage at VEGP is maintained in accordance with this interface agreement. These voltage levels are maintained without any reactive power support from VEGP Units 3 and 4.

VEGP SUP 8.2-4 The agreement between VEGP and SCT demonstrates protocols in place for the plant to remain cognizant of grid vulnerabilities so that they can make informed decisions regarding maintenance activities critical to the electrical system. As part of its operational responsibilities, the PCC continuously monitors real-time power flows and assesses contingency impacts through the use of a state-estimator tool. The PCC/GCC continuously monitors and evaluates grid reliability and switchyard voltages, and informs plant operations of any potential grid instability or voltage inadequacies. They also work to maintain local voltage requirements as required by VEGP. Operational planning studies are also performed using offline power flow study tools to assess near term operating conditions under varying load, generation, and transmission topology patterns. If a condition arises where the SBAA transmission grid cannot supply adequate offsite power, plant operators are notified and appropriate actions are taken.

VEGP plant operations reviews input from the GCC/PCC to make informed decisions regarding plant activities that may affect plant reliability or impacts to the transmission grid. In addition, plant operators inform the PCC/GCC of changes in generation ramp rates and notify them of any developing problems that may impact generation.

VEGP SUP 8.2-2 An agreement between VEGP and SCT sets the requirements for transmission grid studies and analyses. These analyses demonstrate the capability of the offsite power system to support plant start up and shutdown.

VEGP SUP 8.2-3 SCT conducts planning studies of the transmission grid on an ongoing basis. Model data used to perform simulation studies of projected future conditions is maintained and updated as load forecasts and future generation/transmission changes evolve. Studies are updated periodically to assess future system performance in accordance with NERC Reliability Standards. These studies form a basis for identifying future transmission expansion needs.

8.2.1.2 Transformer Area

Add the following paragraph at the end of the first paragraph of DCD Subsection 8.2.1.2.

VEGP COL 8.2-1 The transformer area for each unit contains the main stepup transformer (the GSU), (3 single phase transformers plus one spare), three unit auxiliary transformers (the UATs), and two reserve auxiliary transformers (the RATs). The two RATs are connected to the Units 3 and 4 RAT supply 230 kV switchyard via overhead tie lines, with a portion of Unit 3 RAT "A" supply line being underground between Unit 4 and Unit 3. The secondary windings (230 kV side) of the Unit 3 GSU are connected in a wye configuration and connected to the Units 1, 2 and 3, 230/500 kV switchyard. The secondary windings (500 kV side) of the Unit 4 GSU are connected in a wye configuration and connected to the Unit 4, 500 kV switchyard.

Add the following paragraph and subsections at the end of the DCD Subsection 8.2.1.2.

Each transformer is connected to the switchyard by an offsite circuit beginning at the switchyard side of the breaker(s) within the switchyard and ending at the high voltage terminals of the GSU and RATs.

8.2.1.2.1 Switchyard Protection Relay Scheme

VEGP COL 8.2-2 The switchyards are designed to provide high speed fault clearing while also maintaining high reliability and operational flexibility. The arrangement of the switchyards allow for isolation of components and buses, while preserving VEGP's connection to the grid.

Under normal operating conditions all 230 kV and 500 kV circuit breakers and all bus sectionalizing motor operated disconnect switches are closed and all bus sections are energized.

Each 230 kV and 500 kV transmission line is protected by two independent protection schemes (primary and secondary) to achieve high speed clearing for a fault anywhere on the line and to provide remote back-up protection for remote faults. Each scheme has a pilot protection package and a stand alone step distance line protection package. The breaker failure scheme is initiated by either of the primary or secondary protection schemes and operates through a timing relay, and should a breaker fail to trip within the time setting of its timing relay, the associated breaker failure trip relay will trip and lock out all necessary breakers to isolate the faulted area.

VEGP SUP 8.2-6 The protective devices controlling the switchyard breakers are set with consideration given to preserving the plant grid connection following a turbine trip.

8.2.1.3 Switchyard Control Building

VEGP COL 8.2-1 A separate control building is provided to serve the requirements of each of the three high voltage switchyards. Each control building houses switchyard batteries (redundant battery systems are housed in separate battery rooms and appropriately ventilated) and accommodates a sufficient number of relay/control panels.

The 230 kV and 500 kV switchyard breakers associated with the GSU and RATs are under the functional control of the plant. Transmission line circuit breakers and switches in the switchyards are under the control of the GCC. The 230 kV and 500 kV disconnect switches associated with the GSU and RATs are under the control of the plant. All plant switchyard switching is coordinated between the GCC and the VEGP control room operators.

8.2.1.4 Switchyard and Transmission Lines Testing and Inspection

An agreement between VEGP and SCT for development, maintenance, calibration, testing and modification of transmission lines, switchyards, transformer yards and associated transmission equipment, provides the procedure, policy and organization to carry out maintenance, calibration, testing and inspection of transmission lines and switchyards.

This agreement defines the interfaces and working relationship between VEGP and SCT. As a service to VEGP, SCT performs maintenance, calibration, and testing of VEGP transformer assets at Southern Company nuclear sites. VEGP and SCT are responsible for control of plant/grid interface activities. For reliability, VEGP and SCT coordinate maintenance and testing of off-site power systems. SCT and GPC establish communication and coordination protocols for restoration of external power supply to the nuclear plant on a priority basis.

For performance of maintenance, testing, calibration and inspection, SCT follows its own field test manuals, vendor manuals and drawings, and industry maintenance practices to comply with applicable NERC Reliability Standards.

SCT verifies that these test results demonstrate compliance with design requirements and takes corrective actions as necessary. SCT plans and schedules maintenance activities, notifying the plant and PCC/GCC in advance.

SCT also procures and stores necessary spare parts prior to the commencement of inspection, testing, and maintenance activities.

Transmission lines are currently inspected through an aerial inspection program at least six times per year. Four times per year or more, these inspections focus on conductor, hardware and structure condition assessment. Twice per year, the inspection has a specific focus on right of way encroachments and vegetation management. An integrated vegetation management program, consisting of periodic mechanical mowing or hand clearing, and herbicide application, is used to control vegetation within the boundaries of the transmission line rights of way. Patrols to identify and remove danger trees beyond the formal right-of-way, which could adversely affect the operation of the transmission line, are performed twice per year.

In addition to the aerial patrols, each transmission line structure is inspected every six years by ground crew personnel. These crews perform an in-depth, on-site assessment of the structure, conductor, and hardware. Any maintenance work required is noted and scheduled for a follow-up crew if the inspection crew cannot complete the task while on-site.

The interconnecting switchyard, as well as other substation facilities, has multiple levels of inspection and maintenance, including the following:

- Walk throughs and visual inspections of the entire substation facility.
- Relay functional tests.
- Oil sampling of large power transformers. Oil samples are evaluated through the use of gas chromatography and dielectric breakdown analysis.
- Power circuit breakers are subjected to several levels of inspection and maintenance. The frequency of each is a function of the number of operations and the length of time in service. Maintenance leverages the use of external visual inspection of all functional systems, an external test, and an internal inspection. Frequency of the various maintenance/ inspection efforts is based on a combination of operating history of the type of breaker, industry practice and manufacturer’s recommended maintenance requirements.
- A power factor test (Doble Test) is typically performed on oil filled equipment.
- Thermography is used to identify potential thermal heating issues on buses, conductors, connectors and switches.

8.2.2 GRID STABILITY

Add the following information at the end of DCD Subsection 8.2.2.

VEGP COL 8.2-2 SCT Transmission Planning has performed a transmission system study of the offsite power system. In order to maintain Reactor Coolant Pump (RCP) operation for three seconds following a turbine trip as specified in DCD Subsection 8.2.2, the grid voltage at the high-side of the GSU, and RATs cannot dip more than 0.15 p.u. from the pre-trip steady-state voltage.

The study analyzes transient stability utilizing an appropriate load flow case while considering various fault contingencies. In order to complete the forward looking study, the following assumptions are made:

- Grid voltage is 235 kV and 517 kV
- Unit 3 GSU voltage ratio 230/26 kV with a 1.05 p.u. tap setting
- Unit 4 GSU voltage ratio 525/26 kV with a 1.0 p.u. tap setting
- The 2015 summer off-peak case was used as a starting point for the study. The SBAA load was scaled down to a valley load condition of ~38% of peak and the generation was re-dispatched. Valley load conditions provide the most conservative stability results for nuclear units.

The computer analysis was performed using the Siemens Power Technology International Software PSS/E. The analysis examines two conditions:

- Normal Running
- Turbine Trip

Other conditions (i.e. startup and normal shutdown) are bounded by these analyses.

Table 8.2-201 confirms that the interface requirements for steady state load, inrush kVA for motors, nominal voltage, allowable voltage regulation, nominal frequency, allowable frequency fluctuation, maximum frequency decay rate, and limiting under frequency value for RCP have been met.

VEGP SUP 8.2-4 In addition to turbine trip, the grid stability analysis also considered normally-cleared three-phase faults on the transmission system and three-phase faults followed by breaker failure at the VEGP 500 kV and 230 kV switchyards. A 500 kV line out for maintenance with a normally cleared fault on another 500 kV line was also studied. The results demonstrate that the grid remains stable for the loss of the most critical transmission line, the loss of the largest load, and the loss of the largest generating unit. For these contingencies, the generator bus voltages and switchyard voltages (after fault clearing) remain within acceptable steady state voltage limits.

VEGP SUP 8.2-5 From January 1, 1992 to November 30, 2007, the average grid availability for the two transmission voltages is as follows:

500 kV Transmission Voltage: The West McIntosh and Scherer 500 kV transmission lines for VEGP have an availability of 99.9 % with 25 forced outages. The 500 kV Thomson transmission line is not in service during this time period. The average frequency of forced line outages since 1992 is less than two per year. The leading cause of forced outages of significant duration is lightning and the resulting damage. Other failures of significant line outage duration are substation equipment problems and storms not associated with lightning.

230 kV Transmission Voltage: The five 230 kV transmission lines for VEGP have an availability of 99.6 % with 26 forced outages. The average frequency of forced line outages since 1992 is less than two per year. The leading cause of forced outages of significant duration is a line structure failure (which only occurred once). Other failures of significant line outage duration are substation equipment problems.

8.2.5 COMBINED LICENSE INFORMATION FOR OFFSITE ELECTRICAL POWER

VEGP COL 8.2-1 This COL item is addressed in Subsections 8.2.1, 8.2.1.1, 8.2.1.2, 8.2.1.3 and 8.2.1.4.

VEGP COL 8.2-2 This COL item is addressed in Subsections 8.2.1.2.1 and 8.2.2.

VEGP COL 8.2-2

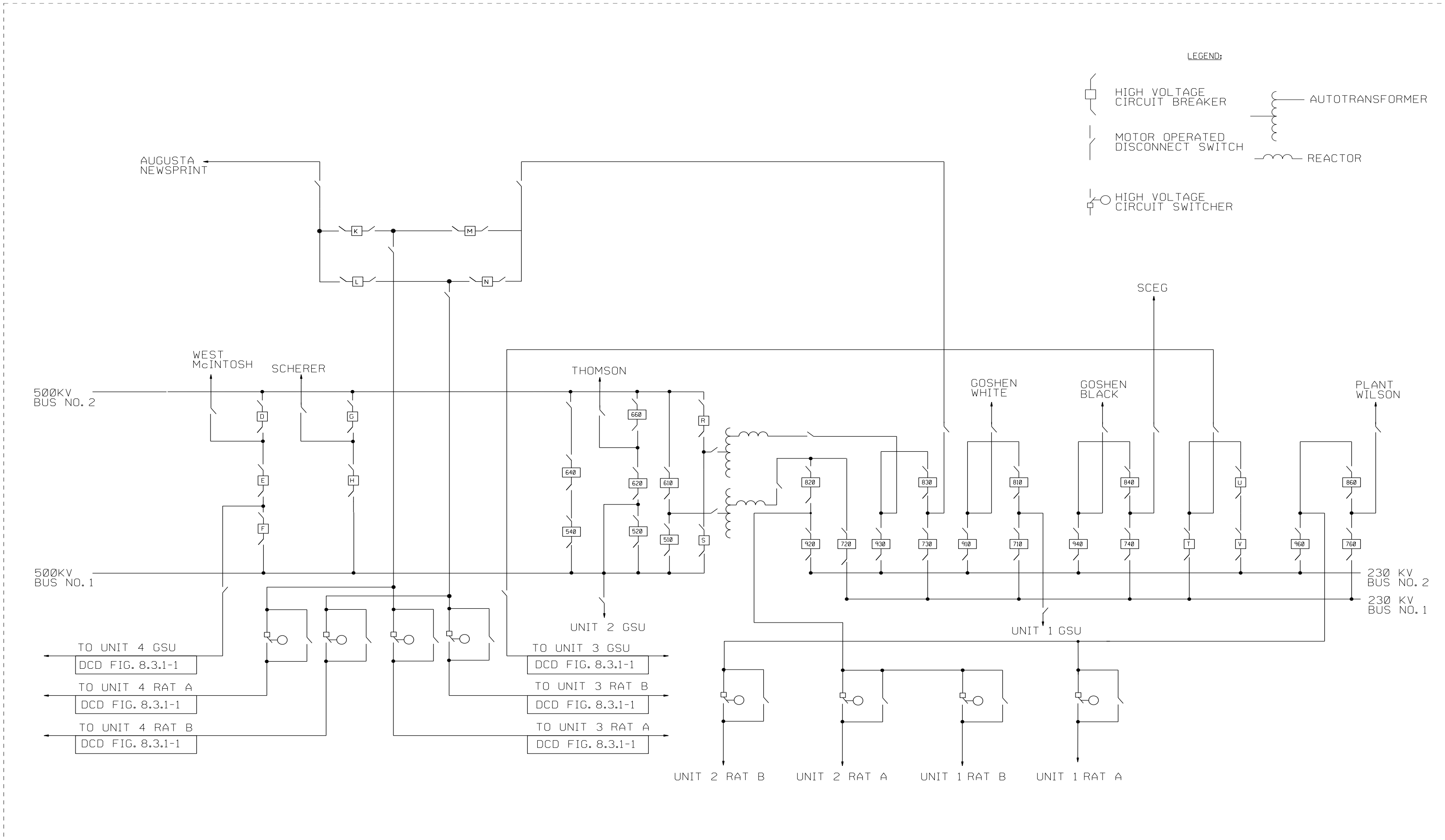
Table 8.2-201
Grid Stability Interface Evaluation

DCD Table 1.8-1 Item 8.2 Parameter	Westinghouse Offsite AC Requirement	VEGP 3 & 4 Value Assumed
Steady-state load	“normal running values provided as input to grid stability”	(78.2 + j 41.7) MVA
Inrush kVA for motors	56,712 kVA*	56,712 kVA @ locked rotor power factor (lrpf) = 0.15 pu
Nominal voltage	Not provided	1.03 pu (517 kV) 1.02 pu (235 kV)
Allowable voltage regulation	0.95–1.05 pu steady state 0.15 pu transient dip**	0.95–1.05 pu steady state 0.15 pu transient dip**
Nominal frequency	60 Hz	60 Hz
Allowable frequency fluctuation	± 1/2 Hz indefinite	± 1/2 Hz indefinite
Maximum frequency decay rate	5 Hz/sec	5 Hz/sec

*Based on the inrush of a single 10,000 HP feedwater pump assuming efficiency = 0.95, pf = 0.9, and inrush = 6.5 x FLA

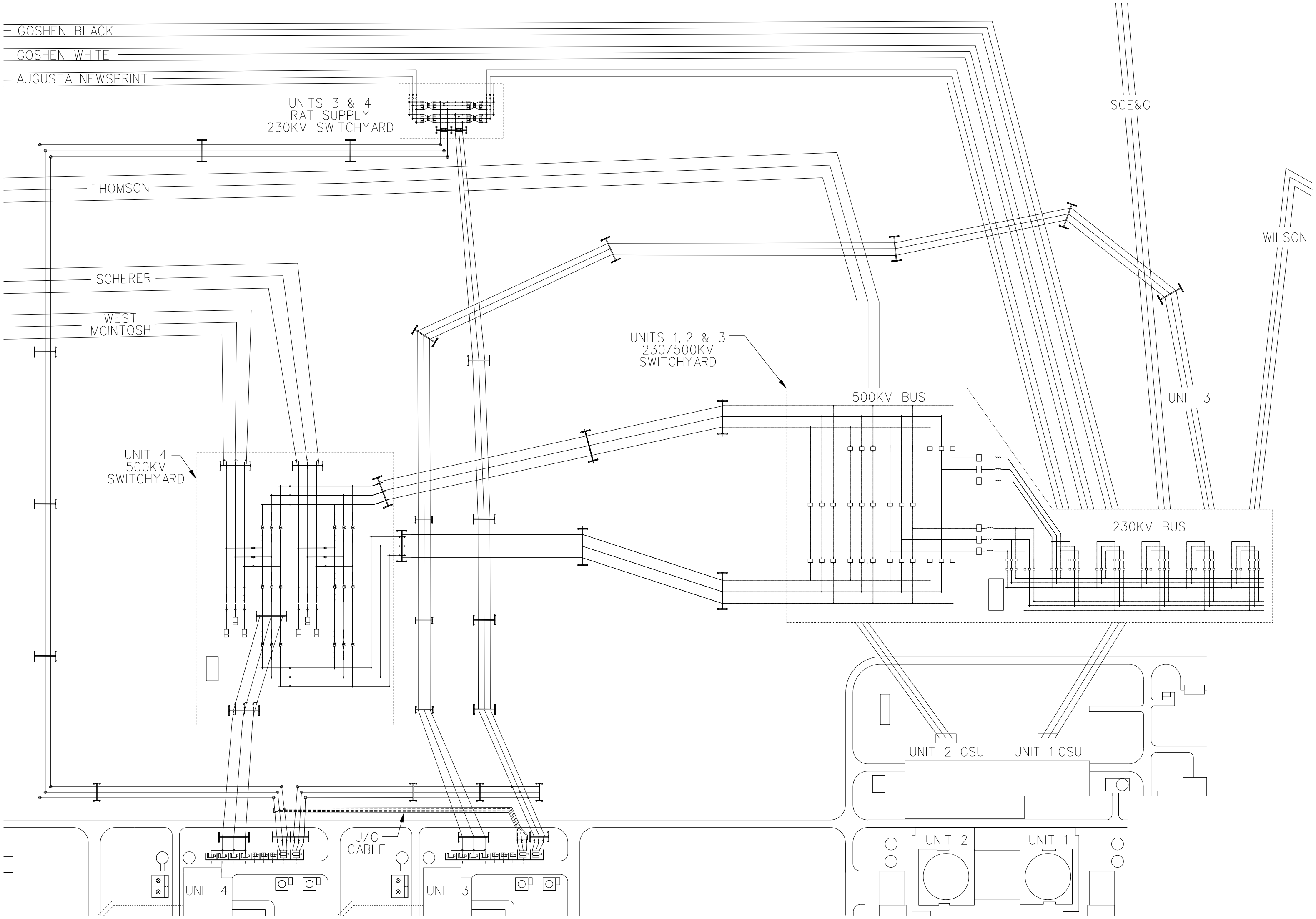
** Applicable to Turbine Trip Only. The maximum allowable voltage dip from the pre-event steady state voltage value during the 3 second turbine trip event transient as measured at the point of connection to the high side of the generator step-up transformer and the reserve auxiliary transformer.

DCD Table 1.8-1 Item 8.2 Parameter	Westinghouse Offsite AC Requirement	VEGP 3 & 4 Value Calculated
Limiting under frequency value for RCP	≥ 57.7 Hz	> 59.85 Hz



VEGP COL 8.2-1

Figure 8.2-201
Offsite Power System One-Line Diagram



VEGP COL 8.2-1

Figure 8.2-202
Switchyard General Arrangement

8.3 ONSITE POWER SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

8.3.1.1.1 Onsite AC Power System

Add the following to the end of the fourth paragraph of DCD Subsection 8.3.1.1.1.

VEGP SUP 8.3-2 The site specific switchyard and transformer voltage is shown on Figure 8.2-201.

8.3.1.1.2.3 Onsite Standby Power System Performance

Add the following text between the second and third paragraphs of DCD Subsection 8.3.1.1.2.3.

VEGP SUP 8.3-1 The VEGP site conditions provided in Section 2.3 are bounded by the standard site conditions used to rate both the diesel engine and the associated generator in DCD Subsection 8.3.1.1.2.3.

Add the following subsection after DCD Subsection 8.3.1.1.2.3.

8.3.1.1.2.4 Operation, Inspection, and Maintenance

STD COL 8.3-2 Operation, inspection and maintenance (including preventive, corrective, and predictive maintenance) procedures consider both the diesel generator manufacturer’s recommendations and industry diesel working group recommendations.

8.3.1.1.6 Containment Building Electrical Penetrations

Add the following text at the end of DCD Subsection 8.3.1.1.6.

STD COL 8.3-2 Procedures implement periodic testing of protective devices that provide penetration overcurrent protection. A sample of each different type of overcurrent device is selected for periodic testing during refueling outages. 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 vacuum circuit breakers.
- Verification of long time, short time, and instantaneous trips of low voltage air circuit breakers.
- Verification of Class 1E and non-Class 1E dc protective device characteristics (except fuses) per manufacturer recommendations, including testing for overcurrent interruption and/or fault current limiting.

Penetration protective devices are maintained and controlled under the plant configuration control program. A fuse control program, including a master fuse list, is established based on industry operating experience.

8.3.1.1.7 Grounding System

Replace the sixth paragraph of DCD Subsection 8.3.1.1.7 with the following information.

VEGP COL 8.3-1 A grounding grid system design within the plant boundary includes step and touch potentials near equipment that 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 80, “IEEE Guide for Safety in AC Substation Grounding” (Reference 201) and a grid configuration for the site was created. The grid configuration was modeled in conjunction with the soil model. The resulting step and touch potentials are within the acceptable limits.

8.3.1.1.8 Lightning Protection

Replace the third paragraph of DCD Subsection 8.3.1.1.8 with the following information.

VEGP COL 8.3-1 In accordance with IEEE 665, "IEEE Standard for Generating Station Grounding" (DCD Section 8.3 Reference 18), a lightning protection risk assessment for the buildings comprising the VEGP Units 3 and 4 was performed based on the methodology in NFPA 780 (DCD Section 8.3 Reference 19). The tolerable lightning frequency for each of the buildings was determined to be less than the expected lightning frequency; therefore, lightning protection is required for the VEGP Units 3 and 4 buildings based on the design in accordance with NFPA 780. The zone of protection is based on the elevations and geometry of the structures. It includes the space covered by a rolling sphere having a radius sufficient enough to cover the building to be protected. The zone of protection method is based on the use of ground masts, air terminals and shield wires. Either copper or aluminum is used for lightning protection. Lightning protection grounding is interconnected with the station or switchyard grounding system.

8.3.1.4 Inspection and Testing

Add the following text at the end of DCD Subsection 8.3.1.4.

STD SUP 8.3-4 Procedures are established 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.

8.3.2.1.1.1 Class 1E DC Distribution

Add the following text at the end of DCD Subsection 8.3.2.1.1.1.

STD SUP 8.3-3 No site-specific non-Class 1E dc loads are connected to the Class 1E dc system.

8.3.2.1.4 Maintenance and Testing

Add the following text at the end of DCD Subsection 8.3.2.1.4.

STD COL 8.3-2 Procedures are established for inspection and maintenance of Class 1E and non-Class 1E batteries. Class 1E battery maintenance and service testing is performed in conformance with Regulatory Guide 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.

The clearing of ground faults on the Class 1E dc system is also addressed by procedure. The battery testing procedures are written in conformance with IEEE 450 and the Technical Specifications.

Procedures are established for periodic testing of the Class 1E battery chargers and Class 1E voltage regulating transformers in accordance with the 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 Maintenance Rule 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.

8.3.2.2 Analysis

Replace the first sentence of the third paragraph of DCD Subsection 8.3.2.2 with the following:

STD DEP 8.3-1 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.

8.3.3 COMBINED LICENSE INFORMATION FOR ONSITE ELECTRICAL POWER

VEGP COL 8.3-1 This COL Item is addressed in Subsections 8.3.1.1.7 and 8.3.1.1.8.

STD COL 8.3-2 This COL Item is addressed in Subsections 8.3.1.1.2.4, 8.3.1.1.6 and 8.3.2.1.4.

8.3.4 REFERENCES

201. Institute of Electrical and Electronics Engineers (IEEE), "IEEE Guide for Safety in AC Substation Grounding," IEEE Std 80-2000, August 4, 2000.

CHAPTER 9

AUXILIARY SYSTEMS

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CHAPTER 9
AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following subsection after DCD Subsection 9.1.4.3.7.

9.1.4.3.8 Radiation Monitoring

STD COL 9.1-6 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 for a discussion of augmented radiation monitoring during fuel handling operations.

9.1.4.4 Inspection and Testing Requirements

Add the following paragraph at the end of DCD Subsection 9.1.4.4.

STD COL 9.1-5 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 light load handling program, including system inspections, is implemented prior to receipt of fuel onsite.

9.1.5 OVERHEAD HEAVY LOAD HANDLING SYSTEMS

Add the following at the end of DCD Subsection 9.1.5.

STD SUP 9.1-2 The heavy loads handling program is based on NUREG 0612 and vendor recommendations. The key elements of the program are:

- Listing of heavy loads to be lifted during operation of the plant. This list will be provided once magnitudes have been accurately formalized but no later than three (3) months prior to fuel receipt.

- Listing of heavy load handling equipment as outlined in DCD Table 9.1-5 and whose characteristics are described in Subsection 9.1.5 of the DCD.

- Heavy load handling safe load paths and routing plans including descriptions of interlocks, (automatic and manual) safety devices and procedures to assure safe load path compliance. Anticipated heavy load movements are analyzed and safe load paths defined. Safe load path considerations are based on comparison with analyzed cases, previously defined safe movement areas, and previously defined restricted areas. The analyses are in accordance with Appendix A of NUREG 0612.

- Heavy load handling equipment maintenance manuals and procedures as described in Subsection 9.1.5.5.

- Heavy load handling equipment inspection and test plans, as outlined in Subsections 9.1.5.4 and 9.1.5.5.

- Heavy load handling personnel qualifications, training, and control procedures as described in Subsection 9.1.5.5.

- QA programs to monitor, implement, and ensure compliance with the heavy load-handling procedures as described in Subsection 9.1.5.5.

A quality assurance program, consistent with Paragraph 10 of NUREG-0554, is established and implemented for the procurement, design, fabrication, installation, inspection, testing, and operation of the crane. The program, as a minimum, includes the following elements:

- design and procurement document control

- instructions, procedures, and drawings

- control of purchased material, equipment, and services

- inspection

- testing and test control

- non-conforming items

- corrective action

- records

9.1.5.3 Safety Evaluation

Add the following information at the end of DCD Subsection 9.1.5.3.

STD SUP 9.1-1 There are no planned heavy load lifts outside those already described in the DCD. However, over the plant life there may be occasions when heavy loads not presently addressed need to be lifted (i.e., in support of special maintenance/ repairs). For these occasions, special procedures are generated that address, as a minimum, the following:

- The special procedure complies with NUREG-0612.
- A safe load path is determined. Mechanical and/or electrical stops are incorporated in the hardware design to prohibit travel outside the safe load path. Maximum lift heights are specified to minimize the impact of an unlikely load drop.
- Where a load drop could occur over irradiated fuel or safe shutdown equipment, the consequence of the load drop is evaluated. If the evaluation concludes that the load drop is not acceptable, an alternate path is evaluated, or the lift is prohibited.
- The lifting equipment is in compliance with applicable ANSI standards and has factors of safety that meet or exceed the requirements of the applicable standards.
- Operator training is provided prior to actual lifts.
- Inspection of crane components is performed in accordance with the manufacturer recommendations.

STD COL 9.1-6 Plant procedures require that an operating radiation monitor is mounted on any crane when it is handling fuel. Refer to DCD Subsection 11.5.6.4 for a discussion of augmented radiation monitoring during fuel handling operations.

9.1.5.4 Inservice Inspection/Inservice Testing

Add the following paragraph at the end of DCD Subsection 9.1.5.4.

STD COL 9.1-5 The above requirements are part of the plant inspection program for the overhead heavy load handling system, which is implemented through procedures. In addition to the above inspections, the procedures reflect the manufacturers' recommendations for inspection and the NUREG-0612 recommendations.

The overhead heavy load handling equipment inservice inspection procedures, as a minimum, address the following:

- Identification of components to be examined
- Examination techniques
- Inspection intervals
- Examination categories and requirements
- Evaluation of examination results

The overhead heavy load handling program, including system inspections, is implemented prior to receipt of fuel onsite.

9.1.5.5 Load Handling Procedures

STD SUP 9.1-3 Load handling operations for heavy loads that are handled over, could be handled over or are in the proximity of irradiated fuel or safe shutdown equipment are controlled by written procedures. As a minimum, procedures are used for handling loads with the spent fuel cask bridge and polar cranes, and for those loads listed in Table 3.1-1 of NUREG 0612. The procedures include and address the following elements:

- The specific equipment required to handle load (e.g., special lifting devices, slings, shackles, turnbuckles, clevises, load cells, etc.).
- Qualification and training of crane operators and riggers in accordance with chapter 2-3.1 of ASME B30.2, "Overhead and Gantry Cranes."
- The requirements for inspection and acceptance criteria prior to load movement.
- The defined safe load path and provisions to provide visual reference to the crane operator and/or signal person of the safe load path envelope.
- Specific steps and proper sequence to be followed for handling load.
- Precautions, limitations, prerequisites, and/or initial conditions associated with movement of heavy loads.

- The testing, inspection, acceptance criteria and maintenance of overhead heavy load handling systems. These procedures are in accordance with the manufacturer recommendations and are consistent with ANSI B30.2 or with other appropriate and applicable ANSI standards.

Safe load paths are defined for movement of heavy loads to minimize the potential for a load drop on irradiated fuel in the reactor vessel, spent fuel pool or safe shutdown equipment. Paths are defined clearly in procedures and equipment layout drawings. Equipment layout drawings showing the safe load path are used to define safe load paths in load handling procedures. Deviation from defined safe load paths requires a written alternative procedure approved by a plant safety review committee.

9.1.6 COMBINED LICENSE INFORMATION FOR FUEL STORAGE AND HANDLING

STD COL 9.1-5 This COL Item is addressed in Subsections 9.1.4.4 and 9.1.5.4.

STD COL 9.1-6 This COL Item is addressed in Subsections 9.1.4.3.8 and 9.1.5.3.

STD COL 9.1-7 A spent fuel rack Metamic coupon monitoring program will be implemented 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. The program will also include testing to monitor changes in physical properties of the absorber material, including neutron attenuation and thickness measurements.

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% 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% 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.

9.2 WATER SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

9.2.1 SERVICE WATER SYSTEM

9.2.1.2.2 Component Description

Add the following paragraph at the end of DCD Subsection 9.2.1.2.2, Component Description, Cooling Tower subsection.

VEGP SUP 9.2-5 The SWS Cooling Tower was evaluated for potential impacts from interference and air restriction effects due to yard equipment layout and cooling tower operation in an adjacent unit. Based on unit spacing, yard equipment layout and the margins inherent in the performance requirements and design conditions of the towers, no adverse impacts were determined.

9.2.5 POTABLE WATER SYSTEM

9.2.5.2.1 General Description

Replace the second paragraph of DCD Subsection 9.2.5.2.1 with the following information.

VEGP COL 9.2-1 The source of water for the potable water system is the site well water subsystem of the Raw Water System (RWS). The potable water system is a common system that consists of a potable water storage tank, two potable water pumps, a jockey pump, and a distribution header that serves the Units 3 and 4 power blocks and associated buildings. The potable water storage tank is located to the south of the CWS cooling towers. The potable water pumps are housed in a pump house near the tank. The system is not shared with the Units 1 and 2 potable water system. The design of this water supply meets or exceeds the pressure, capacity, and quality requirements in the DCD Subsection 9.2.5.

9.2.5.2.2 Component Description

Add the following text to the end of DCD Subsection 9.2.5.2.2.

Potable Water Storage Tank

VEGP COL 9.2-1 The potable water storage tank is sized to provide sufficient potable water to meet system demands. The tank is designed and constructed in accordance with the applicable AWWA and Georgia Environmental Protection Division standards.

Potable Water Pumps

Each of the two motor-driven potable water pumps takes suction from the potable water storage tank and discharges to the domestic water distribution header. The pumps are operated as required to meet the potable water demand in the plant and still maintain a minimum 20 psig pressure at the furthermost point in the distribution system.

Jockey Pump

A continuously operated jockey pump is used to supply potable water to the distribution header and maintain the pressure of the system during periods of low demand. This motor-driven pump takes suction from the potable water storage tank and pumps water through the distribution system. A recirculation line to the potable water storage tank is provided to allow continuous running of the jockey pump when system demand is low.

9.2.5.3 System Operation

Replace the first and second paragraphs of DCD Subsection 9.2.5.3 with the following information.

VEGP DEP 9.2-1
VEGP COL 9.2-1 The RWS well water subsystem provides well water to the potable water storage tank. An RWS fill valve is automatically opened and closed based on potable water storage tank level.

VEGP COL 10.4-3 The well water is disinfected at the potable water storage tank. Sodium hypochlorite is used as the disinfectant. A minimum residual chlorine level of 0.2 ppm is maintained in the system in accordance with Georgia Safe Drinking Water standards.

Two potable water pumps and a system jockey pump are used to supply potable water throughout the system. The potable water system pumps are activated sequentially to maintain the required pressure throughout the distribution system.

A pressure transmitter is provided downstream of the potable water system pumps to control their start/stop sequences. The jockey pump operates continuously to maintain system pressure.

9.2.5.6 Instrumentation Applications

Add the following text to the end of DCD Subsection 9.2.5.6.

VEGP COL 9.2-1 Instrumentation on the potable water storage tank includes level indication for alarm signals and control signals for the fill valve and the potable water system pumps. The potable water system pumps automatically trip on low tank level and automatically restart when level is restored.

Instrumentation is provided to control the feed of disinfectant to maintain adequate residual chlorine levels in the potable water system.

A pressure transmitter located downstream of the potable water system pumps controls the stop/start sequence of the pumps. The jockey pump runs continuously to maintain system pressure. If the jockey pump is unable to maintain system pressure, a potable water system pump is started. The second potable water system pump starts if the first potable water pump cannot maintain acceptable system pressure.

9.2.6 SANITARY DRAINS

Add the following text to the end of DCD Subsection 9.2.6.

VEGP SUP 9.2-1 The SDS transports wastes to the Units 1 and 2 sewage treatment plant for treatment, dilution, and discharge.

9.2.6.2.1 General Description

Add the following to the third paragraph of DCD Subsection 9.2.6.2.1.

VEGP SUP 9.2-1 The site-specific waste treatment plant for VEGP Units 3 and 4 is the Units 1 and 2 sewage treatment plant. The Units 1 and 2 sewage treatment plant has sufficient capacity to process the sanitary waste from all four units.

9.2.6.5 Instrument Application

Add the following to the first paragraph of DCD Subsection 9.2.6.5.

VEGP SUP 9.2-1 The instrumentation associated with the Units 1 and 2 sewage treatment plant is sufficient for operation of the treatment plant with waste streams from all four units.

Subsection 9.2.8 is modified using full text incorporation to provide site-specific information to replace the DCD conceptual design information (CDI).

9.2.8 TURBINE BUILDING CLOSED COOLING WATER SYSTEM

VEGP CDI The turbine building closed cooling water system (TCS) provides chemically treated, demineralized cooling water for the removal of heat from nonsafety-related heat exchangers in the turbine building and rejects the heat to the circulating water system.

9.2.8.1 Design Basis

9.2.8.1.1 Safety Design Basis

DCD The turbine building closed cooling water system has no safety-related function and therefore has no nuclear safety design basis.

9.2.8.1.2 Power Generation Design Basis

The turbine building closed cooling water system provides corrosion-inhibited, demineralized cooling water to the equipment shown in Table 9.2.8-1 during normal plant operation.

VEGP CDI During power operation, the turbine building closed cooling water system provides a continuous supply of cooling water to turbine building equipment at a temperature of 105°F or less assuming a circulating water temperature of 100°F or less.

DCD

The cooling water is treated with a corrosion inhibitor and uses demineralized water for makeup. The system is equipped with a chemical addition tank to add chemicals to the system.

VEGP CDI

The heat sink for the turbine building closed cooling water system is the circulating water system. The heat is transferred to the circulating water through plate type heat exchangers which are components of the turbine building closed cooling water system.

DCD

A surge tank is sized to accommodate thermal expansion and contraction of the fluid due to temperature changes in the system.

One of the turbine building closed cooling system pumps or heat exchangers may be unavailable for operation or isolated for maintenance without impairing the function of the system.

The turbine closed cooling water pumps are provided ac power from the 6900V switchgear bus. The pumps are not required during a loss of normal AC power.

9.2.8.2 System Description

9.2.8.2.1 General Description

VEGP CDI

Classification of equipment and components is given in Section 3.2. 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. Heat is removed from the turbine building closed cooling water system by the circulating water system via the heat exchangers.

DCD

The pumps take suction from a single return header. Either of the two pumps can operate in conjunction with any two of the three heat exchangers. Discharge flows from the heat exchangers combine into a single supply header. Branch lines then distribute the cooling water to the various coolers in the turbine building. The flow rates to the individual coolers are controlled either by flow restricting orifices or by control valves, according to the requirements of the cooled systems. Individual coolers can be locally isolated, where required, to permit maintenance of the cooler while supplying the remaining components with cooling water. A bypass line with a manual valve is provided around the turbine building closed cooling water system heat exchangers to help avoid overcooling of components during startup/low-load conditions or cold weather operation.

The system is kept full of demineralized water by a surge tank which is located at the highest point in the system. The surge tank connects to the system return header upstream of the pumps. The surge tank accommodates thermal expansion and contraction of cooling water resulting from temperature changes in the system. It also accommodates a minor leakage into or out of the system. Water makeup to the surge tank, for initial system filling or to accommodate leakage from the system, is provided by the demineralized water transfer and storage system. The surge tank is vented to the atmosphere.

A line from the pump discharge header bank to the pump suction header contains valves and a chemical addition tank to facilitate mixing chemicals into the closed loop system to inhibit corrosion in piping and components.

A turbine building closed cooling water sample is periodically taken and analyzed to verify that water quality is maintained.

9.2.8.2.2 Component Description

Surge Tank

A surge tank accommodates changes in the cooling water volume due to changes in operating temperature. The tank also temporarily accommodates leakage into or out of the system. The tank is constructed of carbon steel.

Chemical Addition Tank

The chemical addition tank is constructed of carbon steel. The tank is normally isolated from the system and is provided with a hinged closure for addition of chemicals.

Pumps

Two pumps are provided. Either pump provides the pumping capacity for circulation of cooling water throughout the system. The pumps are single stage, horizontal, centrifugal pumps, are constructed of carbon steel, and have flanged suction and discharge nozzles. Each pump is driven by an ac powered induction motor.

Heat Exchangers

Three heat exchangers are arranged in a parallel configuration. Two of the heat exchangers are in use during normal power operation and turbine building closed cooling water flow divides between them.

VEGP CDI

The heat exchangers are plate type heat exchangers. Turbine building closed cooling water circulates through one side of the heat exchangers while circulating water flows through the other side. During system operation, the turbine building closed cooling water in the heat exchangers is maintained at a higher pressure than the circulating water so leakage of circulating water into the closed cooling water system does not occur. The heat exchangers are constructed of titanium plates with a carbon steel frame.

Valves

DCD

Manual isolation valves are provided upstream and downstream of each pump. The pump isolation valves are normally open but may be closed to isolate the non-operating pump and allow maintenance during system operation. Manual isolation valves are provided upstream and downstream of each turbine building closed cooling water heat exchanger. One heat exchanger is isolated from system flow during normal power operation. A manual bypass valve can be opened to bypass flow around the turbine building closed cooling water heat exchanger when necessary to avoid low cooling water supply temperatures.

Flow control valves are provided to restrict or shut off cooling water flow to those cooled components whose function could be impaired by overcooling. The flow control valves are air operated and fail open upon loss of control air or electrical power. An air operated valve is provided to control demineralized makeup water to the surge tank for system filling and for accommodating leakage from the system. The makeup valve fails closed upon loss of control air or electrical power.

A TCS heat exchanger can be taken out of service by closing the inlet isolation valve. Water chemistry in the isolated heat exchanger train is maintained by a continuous flow of circulating water through a small bypass valve around the inlet isolation valve.

Backwashable strainers are provided upstream of each TCS heat exchanger. They are actuated by a timer and have a backup starting sequence initiated by a high differential pressure across each individual strainer. The backwash can be manually activated.

Piping

System piping is made of carbon steel. Piping joints and connections are welded, except where flanged connections are used for accessibility and maintenance of components. Nonmetallic piping may also be used.

9.2.8.2.3 System Operation

The turbine building closed cooling water system operates during normal power operation. The system does not operate with a loss of normal ac power.

Startup

VEGP CDI

The turbine building closed cooling water system is placed in operation during the plant startup sequence after the circulating water system is in operation but prior to the operation of systems that require turbine building closed cooling water flow. The system is filled by the demineralized water transfer and storage system through a fill line to the surge tank. The system is placed in operation by starting one of the pumps.

DCD

Normal Operation

During normal operation, one turbine building closed cooling water system pump and two heat exchangers provide cooling to the components listed in Table 9.2.8-1. The other pump is on standby and aligned to start automatically upon low discharge header pressure.

During normal operation, leakage from the system will be replaced by makeup from the demineralized water transfer and storage system through the automatic makeup valve. Makeup can be controlled either manually or automatically upon reaching low level in the surge tank.

Shutdown

The system is taken out of service during plant shutdown when no longer needed by the components being cooled. The standby pump is taken out of automatic control, and the operating pump is stopped.

9.2.8.3 Safety Evaluation

The turbine building closed cooling water system has no safety-related function and therefore requires no nuclear safety evaluation.

9.2.8.4 Tests and Inspections

Pre-operational testing is described in Chapter 14. The performance, structural, and leaktight integrity of system components is demonstrated by operation of the system.

9.2.8.5 Instrument Applications

Parameters important to system operation are monitored in the main control room. Flow indication is provided for individual cooled components as well as for the total system flow.

Temperature indication is provided for locations upstream and downstream of the turbine building closed cooling water system heat exchangers. High temperature of the cooling water supply alarms in the main control room. Temperature test points are provided at locations to facilitate thermal performance testing.

Pressure indication is provided for the pump suction and discharge headers. Low pressure at the discharge header automatically starts the standby pump.

Level instrumentation on the surge tank provides level indication and both low- and high-level alarms in the main control room. On low tank level, a valve in the makeup water line automatically actuates to provide makeup flow from the demineralized water transfer and storage system.

9.2.9 WASTE WATER SYSTEM

9.2.9.2.1 General Description

Add the following sentence to the fourth paragraph of DCD Subsection 9.2.9.2.1.

VEGP COL 9.2-2 The wastewater retention basin transfer pumps discharge the basin effluent to the blowdown sump which then mixes with the high volume waste stream (circulating water system cooling tower blowdown) prior to discharge to the Savannah River via the outfall piping.

VEGP SUP 9.2-2 Design and routing of the condenser waterbox drains is addressed in Subsection 10.4.5.2.2.

9.2.9.2.2 Component Description

Replace the paragraph in the Waste Water Retention Basin portion of DCD Subsection 9.2.9.2.2 with the following text.

VEGP COL 9.2-2 The waste water retention basin is a lined basin with two compartments and is constructed such that its contents, dissolved or suspended, do not penetrate the liner and leach into the ground. Either of these compartments can receive waste streams for holdup, or if required, for treatment to meet specific environmental discharge requirements.

The configuration and size of the wastewater retention basin allows settling of solids larger than 10 microns. Wastewater can be sampled prior to discharge from the wastewater retention basin. The wastewater retention basins are located northwest of each power block.

Add the following paragraphs at the end of DCD Subsection 9.2.9.2.2.

Basin Transfer Pumps

VEGP COL 9.2-2 Two 100% capacity submersible-type pumps send waste water from the retention basin to the blowdown sump. Each pump is sized to meet the maximum expected influent flow to prevent overflow of the basin. In the event of oily waste leakage into the retention basin, a recirculation line is provided to recycle the oil/water waste from the basin to the oil separator. In the event of radioactive contamination, this same line can be used to send the contents of the basin to the liquid radioactive waste system (WLS). Controls are provided for automatic or manual operation of the pumps based on the level of the retention basin.

Blowdown Sump

VEGP SUP 9.2-3 A blowdown sump common to both Units 3 and 4 receives input from the wastewater retention basins and the circulating water system (CWS) cooling tower blowdown. The blowdown sump is located to the northeast of Units 3 and 4, outside of the protected area. A connection with the river water subsystem of the raw water system provides an alternate dilution source to the blowdown sump. These inputs are mixed with a dechlorination chemical, as needed, to produce an effluent that meets the NPDES permit requirements. This effluent then flows from the blowdown sump to the outfall structure, and then finally to the river.

Plant Outfall

The plant outfall is the final discharge point for Units 3 and 4. The outfall pipe is sized to drain, via gravity, the maximum expected flow from the blowdown sump. Effluent from the blowdown sump mixes with a small waste stream from the liquid radioactive waste system monitor tanks and is discharged eastward to the Savannah River. Dilution water from the raw water system may be supplied to the blowdown sump for radioactive waste discharges when the circulating water cooling tower blowdown is not available. To prevent radioactive contamination of the blowdown sump, the location of the tie-in between the liquid radwaste and the outfall is downstream and below the bottom elevation of the blowdown sump. The liquid radwaste is monitored for radiation and is addressed in detail in DCD Section 11.2; the applicable radiation monitor is addressed in detail in DCD Subsection 11.5.2.3.3.

9.2.9.5 Instrumentation Applications

Add the following at the end of the first paragraph of DCD Subsection 9.2.9.5.

VEGP COL 9.2-2 Level instrumentation is provided at the wastewater retention basin and is used to control operation of the basin transfer pumps. High-level alarms indicate the basin level where operator action is required.

VEGP DEP 1.1-1 Add the following subsection after DCD Subsection 9.2.10. DCD Subsections 9.2.11 and 9.2.12 are renumbered as Subsections 9.2.12 and 9.2.13, respectively.

9.2.11 RAW WATER SYSTEM

VEGP SUP 9.2-4 The raw water system consists of two subsystems.

The RWS river water subsystem provides river water for makeup to the CWS natural draft cooling tower basins. The river water subsystem also provides dilution water to the Units 3 and 4 blowdown sump and fill water for the CWS piping.

The RWS well water subsystem provides well water for makeup to the service water system (SWS) mechanical draft cooling tower basins, potable water system (PWS), fire protection systems (FPS's), yard fire water systems (YFS's), and demineralized water treatment systems (DTS's).

The well water subsystem also provides lubrication and cooling water to the CWS pumps and well water for miscellaneous plant uses.

9.2.11.1 Design Basis

9.2.11.1.1 Safety Design Basis

The RWS serves no safety-related function, and therefore, has no nuclear safety design basis.

Failure of the RWS or its components does not affect the ability of safety-related systems to perform their intended function.

The RWS does not have the potential to be a flow path for radioactive fluids.

9.2.11.1.2 Power Generation Design Basis

9.2.11.1.2.1 Normal Operation

The RWS river water subsystem provides a continuous supply of river water for the following services:

- CWS fill and makeup.
- Dilution water for radwaste discharge when the CWS is not available. Dilution water is provided to the Units 3 and 4 blowdown sump for this purpose.

The RWS well water subsystem provides well water for the following services:

- SWS fill and makeup.
- PWS fill and makeup.
- DTS feed.
- CWS pump lubrication and cooling water.
- Well water for miscellaneous uses, such as equipment washdown.
- Normal filling and makeup to the FPS primary and secondary fire water storage tanks.
- Normal filling and makeup to the YFS fire water storage tanks.

9.2.11.1.2.2 Outage Mode Operation

During plant outages, CWS cooling tower makeup from the river water subsystem of the RWS is not required. The river water subsystem does provide dilution water for radwaste discharge when the CWS is not available. River water is provided to the common Units 3 and 4 blowdown sump for this purpose.

The RWS well water subsystem provides well water to DTS, PWS, FPS, and YFS as needed. CWS pump lubrication and cooling water is typically not required for a unit in outage mode. The RWS well water subsystem provides makeup flow to the SWS to support normal plant cooldown. During this operational sequence, the component cooling water system reduces the temperature of the reactor coolant system from 350° F, at approximately 4 hours after reactor shutdown, to 125° F within 96 hours after shutdown by providing cooling to the normal residual heat removal system (RNS) heat exchangers.

9.2.11.2 System Description

9.2.11.2.1 General Description

Classification of components and equipment for the RWS is given in Section 3.2.

9.2.11.2.1.1 RWS River Water Subsystem

The source of water for the river water subsystem of the RWS is the Savannah River. The river water subsystem is shown in Figure 9.2-201 (Sheet 1).

For the river water subsystem, the water is drawn through an intake canal that extends from the river to an intake structure. Refer to ESPA Subsection 2.4.8.1 for additional description of the intake canal.

The intake structure is divided into a total of nine independent pump bays. Three bays are dedicated to Unit 3, three bays are dedicated to Unit 4, and the last three bays are empty (to be used for a future unit if pursued). Each Unit 3 and 4 bay is equipped with a river water pump, a trash rack, and a traveling screen. Two bays per unit are equipped with screen wash pumps.

The river water pumps draw untreated river water and forward it to the CWS natural draft cooling tower basins. A side connection provides dilution water to the blowdown sump if required. No additional water treatment is required for the RWS river water subsystem.

9.2.11.2.1.2 RWS Well Water Subsystem

Source water for the well water subsystem is from two deep wells located to the south of the CWS cooling towers. The well water subsystem is shown in Figure 9.2-201 (Sheet 2). The quality of the water provided by the deep wells is sufficient for the required services. No additional water treatment is required for the RWS well water subsystem.

The well water makeup pumps supply well water to the common Units 3 and 4 well water storage tank. The well water transfer pumps supply well water from the well water storage tank to the Units 3 and 4 SWS mechanical draft cooling tower basins and the Units 3 and 4 Demineralized Water Treatment Systems (DTS's).

The well water subsystem also provides well water for filling and makeup to the Units 3 and 4 FPS primary and secondary fire water storage tanks and the YFS fire water storage tanks.

The CWS, PWS, and miscellaneous well water users also receive water from the well water subsystem.

9.2.11.2.2 Component Description

9.2.11.2.2.1 RWS River Water Subsystem

Trash Rack

A rack located near the entrance to the pump bays prevents large debris in the river from entering the bays. A trash rake is provided to clear the rack.

Traveling Screens

Traveling screens at the inlet to each of the intake pump bays provide screening of floating and suspended solids in the river water, and minimize entrainment of aquatic life in the water entering each bay. The screens are sized so that the through screen velocity is less than 0.5 feet per second to reduce impingement mortality of aquatic biota. Buildup on the screens is washed off with spray water and sluiced into the river. Each traveling screen is powered by an electric motor powered from the normal AC power system.

Screen Wash Pumps

Two screen wash pumps for each unit provide spray water to remove debris and aquatic biota from the operational traveling screens. One pump operates to provide normal screen wash requirements, with the other pump in standby. The screen wash pumps are vertical shaft, constant speed electric motor-driven pumps. They are powered from the normal AC power system.

River Water Pumps

Three 50 percent river water pumps are available to draw river water and forward it to the CWS natural draft cooling tower basin. Two pumps normally operate to furnish the required normal CWS makeup flow, with one pump in standby. The river water pumps are vertical turbine, constant speed electric motor-driven pumps. They are powered from the normal AC power system.

Piping

The river water piping is designed to accommodate transient effects that may be generated by the starting and stopping of pumps, opening and closing of valves, or other operating events. The system is designed so that high points do not lead to the formation of vapor voids upon loss of system pumping.

Air release valves are provided in the makeup pump discharge piping to vent air on pump start. The underground RWS river water piping is high-density polyethylene (HDPE) pipe, which is not susceptible to corrosion.

The piping is designed and installed in accordance with the ASME B31.1 piping code.

9.2.11.2.2.2 RWS Well Water Subsystem

Makeup Wells

Two deep makeup wells supply water for Units 3 and 4 via well water makeup pumps to a storage tank. The wells are separated by a minimum distance of 1,000 feet and are screened in the Cretaceous aquifer. The Cretaceous aquifer is considered to be confined because of the low permeability of the Blue Bluff Marl. This confinement prevents contamination of the aquifer due to chemical or radiological contamination of the soil or groundwater above the Blue Bluff Marl.

Well Water Tank

The Units 3 and 4 well water tank is supplied by the well water makeup pumps. For normal plant operations, water is provided from this tank via the well water transfer pumps to meet the water demands for both Units 3 and 4. The tank has a minimum capacity of 300,000 gallons and is designed and constructed in accordance with AWWA D100.

Well Water Makeup Pumps

One well water makeup pump is provided in each of the makeup wells. Each well water makeup pump has a minimum capactiy of 1,500 gpm.

During normal operation, one well water makeup pump is capable of supplying sufficient well water to the well water tank to support normal well water demands for both units. Operation of the well water makeup pumps is controlled by well water tank level.

The well water makeup pumps are vertical shaft, constant speed, electric motor-driven well pumps. The pump shaft extends to a sufficient depth in the well to ensure that pump NPSH requirements are met.

The normal power supply to the well water makeup pumps is the offsite retail power system (ZRS).

A dedicated well water pump house package diesel generator provides power to the well water makeup pumps in the event of a loss of the normal AC power supply.

Well Water Transfer Pumps

Four well water transfer pumps are provided downstream of the well water tank. Each well water transfer pump has a minimum capacity of 750 gpm.

During normal operation, one well water transfer pump taking suction from the well water tank provides sufficient well water to meet normal demands of both Units 3 and 4. The other well water transfer pumps will remain in standby.

During periods of high demand, the standby well water transfer pumps will automatically start, as required to support demand. A recirculation line is provided on the pump discharge header to allow water to be recycled to the well water tank, as necessary, to ensure that the well water transfer pumps operate efficiently.

The well water transfer pumps are housed in the well water and PWS pump house. The well water transfer pumps are horizontal shaft, centrifugal, constant speed, electric motor-driven pumps.

The normal power supply to the well water transfer pumps is the offsite retail power system (ZRS). In the event of a loss of the normal AC power supply, the well water transfer pumps are powered by the dedicated well water pump house package diesel generator.

Piping

The well water piping is designed to accommodate transient effects that may be generated by the starting and stopping of pumps, opening and closing of valves, or other operating events.

The system is designed so that high points do not lead to the formation of vapor voids upon loss of system pumping. Air release valves are provided in the well water makeup pump discharge piping to vent air on pump start.

The underground RWS well water piping is high-density polyethylene (HDPE) pipe, which is not susceptible to corrosion. The piping is designed and installed in accordance with the ASME B31.1 piping code. Heat tracing has been provided on above ground pipe lines that are susceptible to freezing.

Valves

Most of the valves in the RWS well water subsystem are manual valves used to isolate the various well water demands or to isolate well water subsystem equipment or components for maintenance.

Two motor-operated valves are provided, one on each of the recirculation lines from the well water transfer pump discharge header to the well water tank. The normal power supply to these valves is the offsite retail power system (ZRS). In the event of a loss of the normal AC power supply, the valves will be powered by the dedicated well water pump house package diesel generator. The valves are also equipped with handwheels to allow for manual operation if required.

Well Water Pump House Diesel Generator

A dedicated diesel generator is located at the well water pump house. This diesel generator provides backup AC power to the well water makeup pumps, the well water transfer pumps and associated well water system auxiliary equipment and controls required to support continued operation of the well water system if normal power is not available. The diesel generator also supplies backup power to the potable water system to meet state requirements. The capacity of the diesel generator is sufficient to support continuous well water makeup to the Units 3 and 4 SWS cooling tower basins and the Units 3 and 4 potable water system.

9.2.11.3 System Operation

9.2.11.3.1 RWS River Water Subsystem

The RWS river water subsystem operates during normal modes of operation, including startup, power operation, cooldown, shutdown, and refueling.

9.2.11.3.1.1 Plant Startup

The RWS river water subsystem provides water to the CWS cooling tower basin to fill the CWS piping and to replace evaporative losses as the CWS cooling tower is placed into operation.

9.2.11.3.1.2 Power Operation

During normal operation the river water pumps provide water to the CWS basin to replace CWS cooling tower water losses due to evaporation, drift and blowdown.

9.2.11.3.1.3 Plant Cooldown/Shutdown/Refueling

RWS river water makeup to the CWS cooling tower basin is terminated with the shutdown of the CWS after the turbine is taken off-line and a condenser vacuum is no longer required. However, the system is operated as required to support dilution of plant liquid discharges during these modes.

9.2.11.3.1.4 Loss of Normal AC Power Operation

The RWS river water pumps are not available or required during a loss of offsite power (LOOP) when the main turbine is tripped.

9.2.11.3.2 RWS Well Water Subsystem

The RWS well water subsystem operates to meet varying well water demands during normal modes of operation, including startup, power operation, cooldown, shutdown, and refueling.

9.2.11.3.2.1 Plant Startup

The RWS well water subsystem provides well water to the SWS cooling tower basin to initially fill the basin and piping, and to provide makeup to replace evaporative losses as the SWS is placed into operation.

The RWS well water subsystem also supplies makeup water to FPS, DTS, YFS, and PWS on demand.

Well water is also supplied to the CWS for bearing lubrication and cooling of the CWS pumps when those pumps are placed into operation.

9.2.11.3.2.2 Power Operation

During normal power operation, the RWS well water subsystem provides makeup to the SWS cooling tower basin to replace water losses due to evaporation, drift, and blowdown, and also provides a continuous supply of water to the CWS pumps for bearing lubrication and cooling.

Well water is also supplied to the FPS, DTS, YFS, and PWS systems on demand.

9.2.11.3.2.3 Plant Cooldown/Shutdown

The RWS well water subsystem alignment during plant cooldown/shutdown uses the same alignment as with normal plant operation. The demand for well water makeup to the SWS cooling tower basin is greater due to added shutdown cooling demand on the SWS system.

CWS pump bearing lubrication and cooling well water demands may be secured for the unit(s) in cooldown/shutdown once the CWS pumps have been shutdown.

Well water is also supplied to the FPS, DTS, YFS, and PWS systems on demand.

9.2.11.3.2.4 Refueling

During refueling, the RWS well water subsystem provides makeup to the SWS cooling tower basin to replace SWS cooling tower water losses due to evaporation, drift, and blowdown.

Well water is also supplied to the FPS, DTS, YFS, and PWS systems on demand.

Well water for CWS pump bearing lubrication and cooling is normally not required for unit(s) in this mode.

9.2.11.3.2.5 Loss of Normal AC Power Operation

Backup AC power to the well water makeup pumps, the well water transfer pumps and associated instrumentation and controls is provided from the well water pump house diesel generator. Therefore, the well water system is available for operation during a loss of normal power to provide well water makeup to the SWS cooling tower basins. Only one well water makeup pump and one well water transfer pump are required to support well water makeup to both units SWS during a loss of normal AC power.

The flow control valve on the makeup line to the service water cooling towers is part of the service water system which is described in DCD Subsection 9.2.1. This valve isolates on a loss of normal AC power.

Manual action is required during a loss of normal power to align this valve to provide well water makeup to the SWS cooling tower basin. During a loss of normal power, the well water subsystem will re-circulate water to the well water tank until the makeup flow path is restored by manual alignment.

The well water subsystem is designed to prevent transient water hammer associated with the re-start of system pumps following a loss of normal power. The majority of the system piping is buried below grade, whereas the system demand points and the well water tank are located above grade. Therefore, drainage of large sections of pipe is precluded. In addition, the well water subsystem is equipped with check valves and air release valves as required to prevent the formation of voids within the piping.

9.2.11.4 Safety Evaluation

The RWS has no safety-related function, and therefore, requires no nuclear safety evaluation.

The RWS does not have the potential to be a flow path for radioactive fluids. The RWS has no direct interconnection with any system that contains radioactive fluids.

The RWS river water subsystem supplies an alternate source of dilution for radwaste discharge when the CWS is not in use. To accomplish this, river water is routed to the WWS blowdown sump. The WWS blowdown sump is located on the bluff above the Savannah River. The blowdown sump discharges to the outfall pipe, which in turn drains by gravity into the Savannah River. The radwaste discharge (WLS) line connects to the WWS outfall pipe at a location near the river. The connection between the WWS outfall pipe and the WLS line is located downstream of and more than 100 feet below the blowdown sump, thereby eliminating the ability to contaminate RWS from WLS effluent.

9.2.11.5 Tests and Inspections

Initial test requirements for the RWS are described in Subsection 14.2.9.4.24. System performance and structural and pressure integrity of system components are demonstrated by operation of the system, monitoring of system parameters such as flow, level and pressure, and visual inspections.

9.2.11.6 Instrumentation Applications

9.2.11.6.1 RWS River Water Subsystem

Level indication at the CWS cooling tower basin is used to control the makeup valve to maintain the appropriate water level in the basin.

Level indication is provided in the river intake structure pump bays.

Flow indication is provided for the RWS river water pipe lines to the individual CWS cooling tower basins.

Pressure indication is provided for the RWS river water pump discharge and header piping.

9.2.11.6.2 RWS Well Water Subsystem

Level instrumentation is provided at the well water storage tank and will allow the tank level to be monitored. High and low level signals provide inputs to the well water makeup pump control logic. A low level signal will start one well water makeup pump. A low-low level signal will start the standby well water makeup pump and will initiate an alarm in the control room. A low-low-low level signal will trip the well water transfer pumps and will generate an alarm in the control room. A high level signal will stop the well water makeup pumps. A high-high level signal will generate an alarm in the control room.

Flow instrumentation is provided at the discharge of each well water makeup pump and will allow the flow to be monitored. Low flow on an operating pump will stop the pump and provide an alarm to the control room. Flow instrumentation is also provided for each well water transfer pump and will allow well water transfer pump flow to be monitored.

Pressure instrumentation is provided at the well water transfer pump discharge header and will allow the pressure to be monitored. Normally, one well water transfer pump is run continuously. A low pressure signal will cause startup of additional well water transfer pumps with sequencing in accordance with the control logic. A high pressure signal will cause well water transfer pumps to stop in sequence according to the control logic. Control room alarms are generated for high-high pressure or low-low pressure. Local pressure indication is also provided on each well water transfer pump discharge line.

VEGP DEP 1.1-1

9.2.12 COMBINED LICENSE INFORMATION

9.2.12.1 Potable Water

VEGP COL 9.2-1

This COL item is addressed in Subsections 9.2.5.2.1, 9.2.5.2.2, 9.2.5.3, and 9.2.5.6.

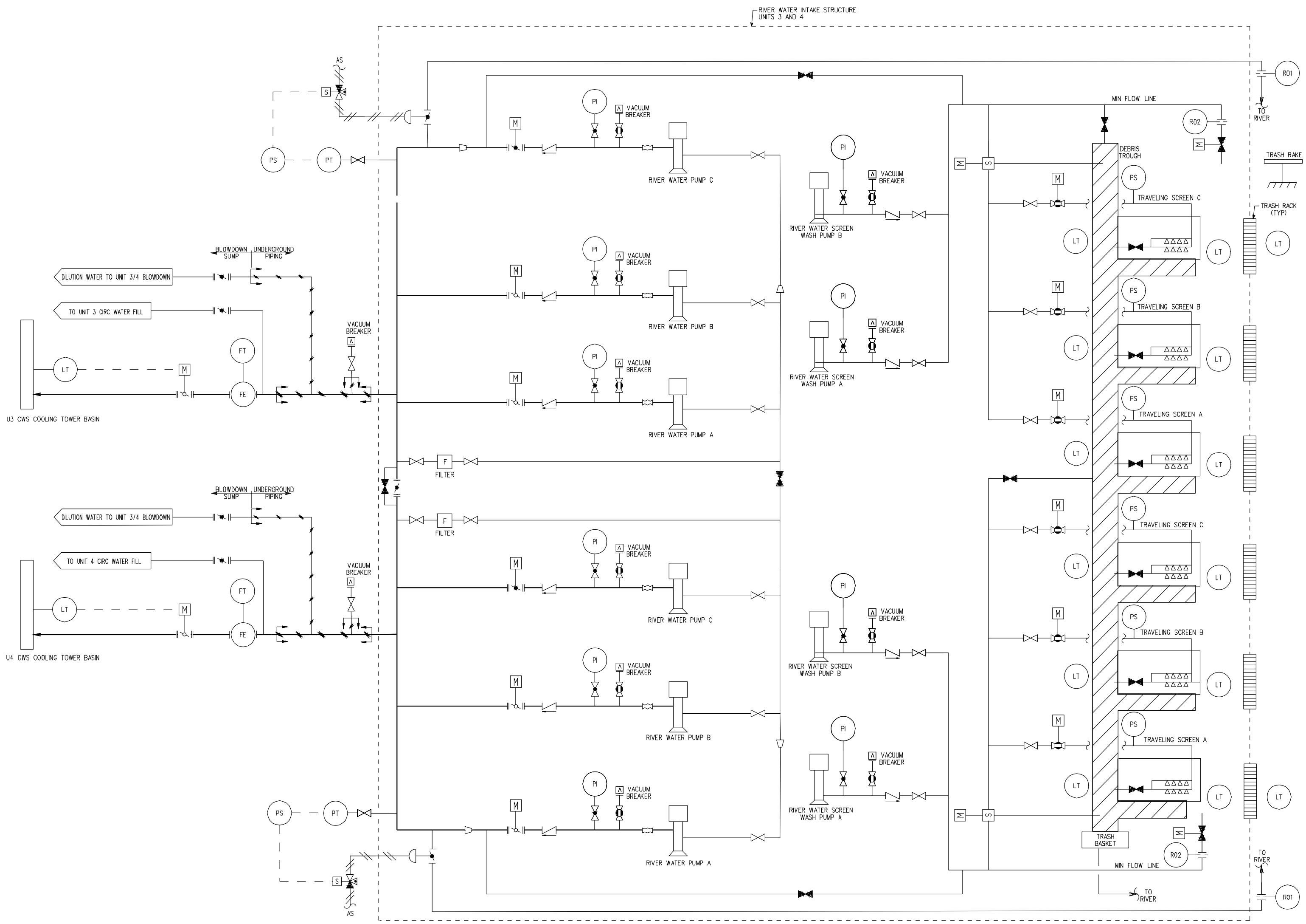
9.2.12.2 Waste Water Retention Basins

VEGP COL 9.2-2

This COL item is addressed in Subsections 9.2.9.2.1, 9.2.9.2.2, and 9.2.9.5.

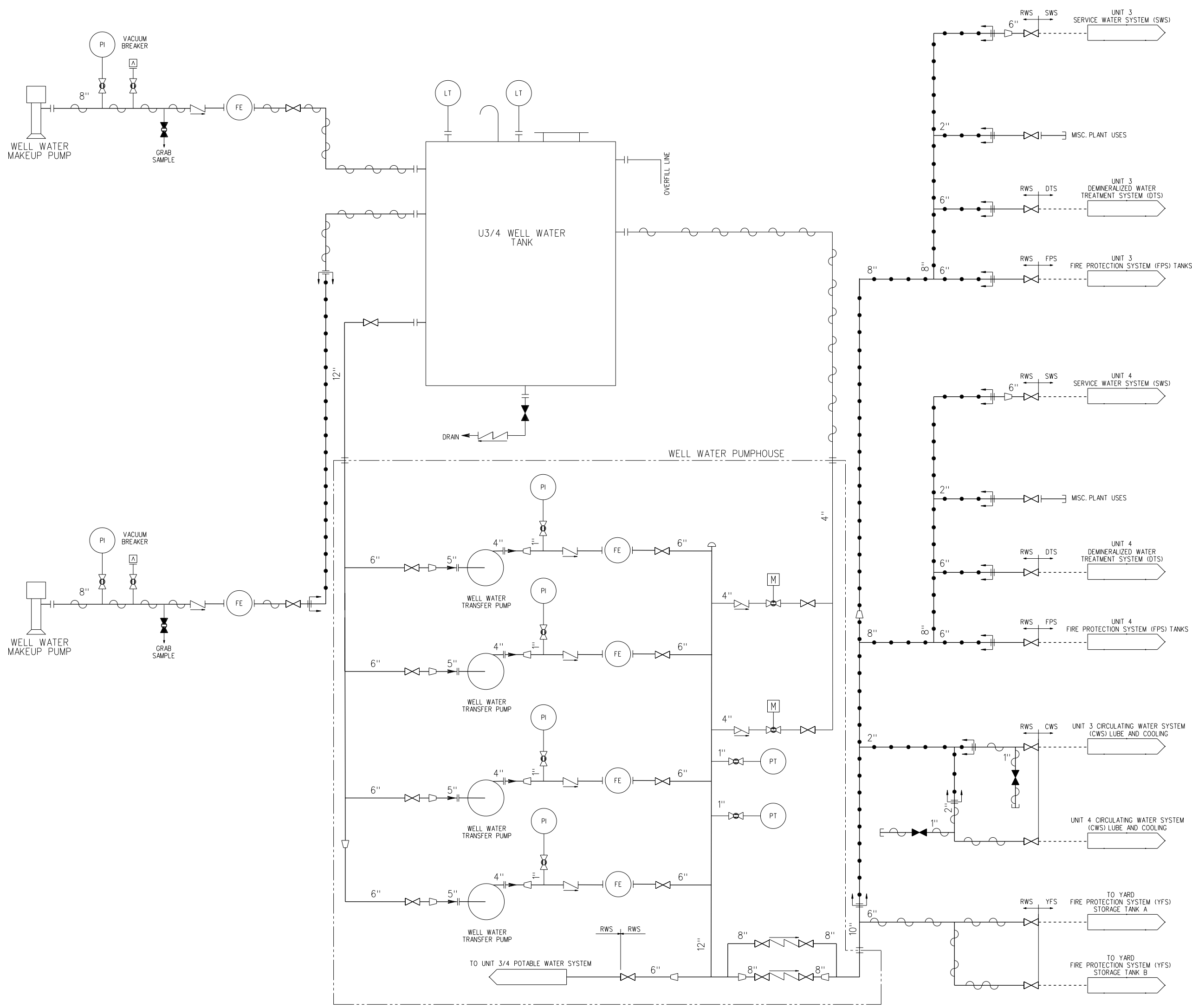
VEGP DEP 1.1-1

9.2.13 REFERENCES



VEGP COL 9.2-1

Figure 9.2-201
Raw Water System River Water Subsystem (Sheet 1 of 2)



VEGP COL 9.2-1

Figure 9.2-201
Raw Water System Well Water Subsystem (Sheet 2 of 2)

9.3 PROCESS AUXILIARIES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements:

9.3.7 COMBINED LICENSE INFORMATION

STD COL 9.3-1 This COL Item is addressed below.

Generic Issue 43, and the concerns of Generic Letter 88-14 and NUREG-1275 regarding degradation or malfunction of instrument air supply and safety-related valve failure, are addressed by the training and procedures for operations and maintenance of the instrument air subsystem and air-operated valves.

Plant systems, including the compressed and instrument air system, are maintained in accordance with procedures . Maintenance procedures are discussed in Subsection 13.5.2.2.6. The instrument air supply subsystem components are maintained and tested in accordance with manufacturers' recommendations and procedures. The safety-related air-operated valves are maintained in accordance with manufacturers' recommendations and tested in accordance with plant procedures to allow proper function on loss of air. The instrument air is periodically sampled and tested for compliance with the quality requirements of ANSI/ISA-S7.3-1981.

Operators are provided training on loss of instrument air in accordance with abnormal operating procedures. Plant systems, including the compressed and instrument air system, are operated in accordance with system operating procedures, abnormal operating procedures, and alarm response procedures which are written in accordance with Subsection 13.5.2. The training program for operations and maintenance personnel is discussed in Section 13.2.

9.4 AIR-CONDITIONING, HEATING, COOLING, AND VENTILATION SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

9.4.1.4 Tests and Inspection

Add the following text at the end of DCD subsection 9.4.1.4.

STD COL 9.4-1a The main control room/control support area HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS) is tested and inspected in accordance with ASME/ANSI AG-1-1997 and Addenda AG-1a-2000 (Reference 201), ASME N509-1989, ASME N510-1989, and Regulatory Guide 1.140.

The VBS is tested as separate components and as an integrated system. Surveillance tests are performed to monitor the condition of the system. Testing methods include:

- Visual inspection
- Duct and housing leak tests
- Airflow capacity and distribution tests
- Air-aerosol mixing uniformity test
- HEPA filter bank and adsorber bank in-place leak tests
- Duct damper bypass tests
- System bypass tests
- Air heater performance tests
- Laboratory testing of adsorbers
- Ductwork inleakage test

Testing is performed at the frequency provided in Table 1 of ASME N510-1989.

9.4.7.4 Tests and Inspections

Add the following text at the end of DCD Subsection 9.4.7.4.

STD COL 9.4-1a The exhaust subsystem of the containment air filtration system (VFS) is tested and inspected in accordance with ASME/ANSI AG-1-1997 and Addenda AG-1a-2000 (Reference 201), ASME N509-1989, ASME N510-1989, and Regulatory Guide 1.140.

The VFS is tested as separate components and as an integrated system. Surveillance tests are performed to monitor the condition of the system. Testing methods include:

- Visual inspection
- Airflow capacity and distribution tests
- HEPA filter bank and adsorber bank in-place leak tests
- System bypass tests
- Air heater performance tests
- Laboratory testing of adsorbers
- Ductwork inleakage test

Testing is performed at the frequency provided in Table 1 of ASME N510-1989.

9.4.12 COMBINED LICENSE INFORMATION

STD COL 9.4-1a This COL Item is addressed in Subsections 9.4.1.4 and 9.4.7.4.

VEGP COL 9.4-1b Section 6.4 does not identify any toxic emergencies that require the main control room/control support area HVAC to enter recirculation mode.

9.4.13 REFERENCES

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201. ASME/ANSI AG-1a-2000, Addenda to ASME AG-1-1997 Code on Nuclear Air and Gas Treatment, Section HA, “Housings.”

9.5 OTHER AUXILIARY SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

9.5.1.2.1.3 Fire Water Supply System

Add the following paragraph at the end of DCD Subsection 9.5.1.2.1.3.

STD SUP 9.5-1 Threads compatible with those used by the off-site fire department are provided on all hydrants, hose couplings and standpipe risers, or a sufficient number of thread adapters compatible with the off-site fire department are provided.

9.5.1.6 Personnel Qualification and Training

Add the following paragraph at the end of DCD Subsection 9.5.1.6.

STD COL 9.5-1 Subsections 9.5.1.8.2 and 9.5.1.8.7 summarize the qualification and training programs that are established and implemented for the Fire Protection Program.

VEGP DEP 1.1-1 Insert the following subsections after DCD Subsection 9.5.1.7. DCD Subsection 9.5.1.8 is renumbered as Subsection 9.5.1.9

9.5.1.8 Fire Protection Program

STD COL 9.5-1 The fire protection program is established such that a fire does not prevent safe shutdown of the plant and does not endanger the health and safety of the public. Fire protection at the plant uses a defense-in-depth concept that includes fire prevention, detection, control and extinguishing systems and equipment, administrative controls and procedures, and trained personnel. These defense-in-depth principles are achieved by meeting the following objectives:

- Prevent fires from starting.
- Detect rapidly, control, and extinguish promptly those fires that do occur.

- Provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities does not prevent the safe shutdown of the plant.

- Minimize the potential for radiological releases.

9.5.1.8.1 Fire Protection Program Implementation

As indicated in Table 13.4-201, the required elements of the fire protection program are fully operational prior to receipt of new fuel for buildings storing new fuel and adjacent fire areas that could affect the fuel storage area in that reactor unit. Other required elements of the fire protection program described in this section are fully operational prior to initial fuel loading in that reactor unit.

Elements of the fire protection program are reviewed on a frequency established by procedures and updated as necessary.

9.5.1.8.1.1 Fire Protection Program Criteria

STD COL 9.5-3 The fire protection program is based on the criteria of several industry and regulatory documents referenced in FSAR Subsection 9.5.5 and DCD Subsection 9.5.5, and also based on the guidance provided in Regulatory Guide 1.189. DCD Tables 9.5.1-1 and FSAR Table 9.5-201 provide a cross-reference to information addressing compliance with BTP CMEB 9.5-1.

STD COL 9.5-4 Exceptions to the National Fire Protection Association (NFPA) Standards beyond those included in DCD Table 9.5.1-3, and exceptions taken to the NFPA Standards listed in FSAR Subsection 9.5.5, are identified in FSAR Table 9.5-202.

9.5.1.8.1.2 Organization and Responsibilities

STD COL 9.5-1 The organizational structure of the fire protection personnel is discussed in Subsection 13.1.1.2.10.

The site executive in charge of the fire protection program, through the engineer in charge of fire protection, is responsible for the following:

a. Programs and periodic inspections are implemented to:

    1. Minimize the amount of combustibles in safety-related areas.

    2. Determine the effectiveness of housekeeping practices.

    3. Provide for availability and acceptability of the following:

i. Fire protection system and components.

ii. Manual firefighting equipment.

iii. Emergency breathing apparatus.

iv. Emergency lighting.

v. Portable communication equipment.

STD COL 9.5-8

STD COL 9.5-1

vi. Fire barriers including fire rated walls, floors and ceilings, fire rated doors, dampers, etc., fire stops and wraps, and fire retardant coating. Procedures address the administrative controls in place, including fire watches, when a fire area is breached for maintenance.

STD COL 9.5-1

4. Confirm prompt and effective corrective actions are taken to correct conditions adverse to fire protection and preclude their recurrence.

b. Conducting periodic maintenance and testing of fire protection systems, components, and manual firefighting equipment, evaluating test results, and determining the acceptability of systems under test in accordance with established plant procedures.

c. Designing and selecting equipment related to fire protection.

d. Reviewing and evaluating proposed work activities to identify potential transient fire loads.

e. Managing the plant fire brigade, including:

1. Developing, implementing, and administering the fire brigade training program.

2. Scheduling and conducting fire brigade drills.

3. Critiquing fire drills to determine if training objectives are met.

4. Performing a periodic review of the fire brigade roster and initiating changes as needed.

5. Maintaining the fire training program records for members of the fire brigade and other personnel.

6. Maintaining a sufficient number of qualified fire brigade personnel to respond to fire emergencies for each shift.

f. Developing and conducting the fire extinguisher training program.

g. Implementing a program for indoctrination of personnel gaining unescorted access to the protected area in appropriate procedures which implement the fire protection program, such as fire prevention and fire reporting procedures, plant emergency alarms, including evacuation.

h. Implementing a program for instruction of personnel on the proper handling of accidental events such as leaks or spills of flammable materials.

i. Preparing procedures to meet possible fire situations in the plant and for ensuring assistance is available for fighting fires in radiological areas.

j. Implementing a program that uses a permit system that controls and documents inoperability of fire protection systems and equipment. This program initiates proper notifications and compensatory actions, such as fire watches, when inoperability of any fire protection system or component is identified.

k. Developing and implementing preventive maintenance, corrective maintenance, and surveillance test fire protection procedures.

l. Confirming that plant modifications, new procedures and revisions to procedures associated with fire protection equipment and systems that have significant impact on the fire protection program, are reviewed by an individual who possesses the qualifications of a fire protection engineer.

m. Continuing evaluation of fire hazards during construction or modification of other units on the site. Special considerations, such as fire barriers, fire protection capability, and administrative controls are provided as necessary to protect the operating unit(s) from construction or modification activities.

n. Establishing a fire prevention surveillance plan and training plant personnel on that plan.

o. Developing prefire plans and making them available to the fire brigade and control room.

The responsibilities of the engineer in charge of fire protection and his staff are discussed in Subsection 13.1.1.2.10.

9.5.1.8.2 Fire Brigade

9.5.1.8.2.1 General

The organization of the fire brigade is discussed in Subsection 13.1.2.1.5.

To qualify as a member of the fire brigade, an individual must meet the following criteria:

a. Has attended the required training sessions for the position occupied on the fire brigade.

b. Has passed an annual physical exam including demonstrating the ability for performing strenuous activity and the use of respiratory protection.

9.5.1.8.2.2 Fire Brigade Training

A training program is established so that the capability to fight fires is developed and documented. The program consists of classroom instruction supplemented with periodic classroom retraining, practice in firefighting, and fire drills. Classroom instruction and training is conducted by qualified individuals knowledgeable in fighting the types of fires that could occur within the plant and its environs and using onsite firefighting equipment. Individual records of training provided to each fire brigade member, including drill critiques, are maintained as part of the permanent plant files for at least three years to document that each member receives the required training.

The fire brigade leader and at least two brigade members per shift have sufficient training and knowledge of plant safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability. The brigade leader is competent to assess the potential safety consequences of a fire and advise control room personnel. Such competence by the brigade leader may be evidenced by possession of an operator’s license or equivalent knowledge of plant systems.

Personnel assigned as fire brigade members receive formal training prior to assuming brigade duties. The course subject matter is selected to satisfy the requirements of Regulatory Guide 1.189. Course material selection also includes guidance from NFPA 600 (Reference 204) and 1500 (Reference 210) as appropriate. Additional training may also include material selected from NFPA 1404 (Reference 208) and 1410 (Reference 209).

The minimum equipment provided for the fire brigade consists of personal protective equipment such as turnout coats, boots, gloves, hard hats, emergency communications equipment, portable lights, portable ventilation equipment, and portable extinguishers. Self-contained breathing apparatus (SCBA) approved by NIOSH, using full face positive pressure masks, and providing an operating life of at least 30 minutes, are provided for selected fire brigade, emergency repair, and control room personnel. At least ten masks are provided for fire brigade personnel. At least two extra air bottles, each with at least 30 minutes of operating life, are located onsite for each SCBA. An additional onsite 6-hour supply of reserve air is provided to permit quick and complete replenishment of exhausted supply air bottles. DCD Subsection 6.4.2.3 discusses the portable breathing apparatus for control room personnel. Additional SCBAs are provided near the personnel containment entrance for the exclusive use of the fire brigade. The fire brigade leader has ready access to keys for any locked fire doors.

The on-duty shift manager has responsibility for taking certain actions based on an assessment of the magnitude of the fire emergency. These actions include safely shutting down the plant, making recommendations for implementing the Emergency Plan, notification of emergency personnel, and requesting assistance from off-duty personnel, if necessary. Emergency Plan consideration of fire emergencies includes the guidance of Regulatory Guide 1.101.

9.5.1.8.2.2.1 Classroom Instruction

Fire brigade members receive classroom instruction in fire protection and firefighting techniques prior to qualifying as members of the fire brigade. This instruction includes:

a. Identification of the types of fire hazards along with their location within the plant and its environs.

b. Identification of the types of fires that could occur within the plant and its environs.

c. Identification of the location of onsite fire fighting equipment and familiarization with the layout of the plant including ingress and egress routes to each area.

d. The proper use of onsite fire fighting equipment and the correct method of fighting various types of fires including at least the following:

   - fires involving radioactive materials
   - fires in energized electrical equipment
   - fires in cables and cable trays
   - fires involving hydrogen
   - fires involving flammable and combustible liquids or hazardous process chemicals
   - fires resulting from construction or modifications (welding)

- fires involving record files.

e. Review of each individual's responsibilities under the Fire Protection Program.

f. Proper use of communication, lighting, ventilation, and emergency breathing equipment.

g. Fire brigade leader direction and coordination of firefighting activities.

h. Toxic and radiological characteristics of expected combustion products.

i. Proper methods of fighting fires inside buildings and confined spaces.

j. Detailed review of firefighting strategies, procedures and procedure changes.

k. Indoctrination of the plant firefighting plans, identification of each individual's responsibilities, and review of changes in the firefighting plans resulting from fire protection-related plant modifications.

l. Coordination between the fire brigade and offsite fire departments that have agreed to assist during a major fire onsite is provided to establish responsibilities and duties. Educating the offsite organization in operational precautions when fighting fires on nuclear power plant sites, and awareness of special hazards and the need of radiological protection of personnel.

9.5.1.8.2.2.2 Retraining

Classroom refresher training is scheduled on a biennial basis to supplement retention of the initial training. These sessions may be concurrent with the regular planned meetings.

9.5.1.8.2.2.3 Practice

Practice sessions are held for each fire brigade and for each fire brigade member on the proper method of fighting various types of fires which might occur in the plant. These sessions are scheduled on an annual basis and provide brigade members with team experience in actual fire fighting and the use of emergency breathing apparatus under strenuous conditions encountered in fire fighting.

9.5.1.8.2.2.4 Drills

Fire brigade drills are conducted at least once per calendar quarter for each shift. Each fire brigade member participates in at least two drills annually. Drills are either announced or unannounced. At least one unannounced drill is held annually for each shift 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. Drills are critiqued on the following points:

a. Assessment of fire alarm effectiveness.

b. Assessment of time required to notify and assemble the fire brigade.

c. Assessment of the selection, placement, and use of equipment.

d. Assessment of the fire brigade leader's effectiveness in directing the firefighting effort.

e. Assessment of each fire brigade member's knowledge of firefighting strategy, procedures, and simulated use of equipment.

f. Assessment of the fire brigade's performance as a team.

Performance deficiencies identified, based on these assessments, are used as the basis for additional training and repeat drills. Unsatisfactory drill performance is followed by a repeat drill within 30 days.

9.5.1.8.2.2.5 Meetings

Regular planned meetings are held at least quarterly for the fire brigade members to review changes in the Fire Protection Program and other subjects as necessary.

9.5.1.8.3 Administrative Controls

Administrative controls for the Fire Protection Program are implemented through plant administrative procedures. Applicable industry publications are used as guidance in developing those procedures.

Administrative controls include procedures to:

a. Control actions to be taken by an individual discovering a fire, such as notification of the control room, attempting to extinguish the fire, and actuation of local fire suppression systems.

b. Control actions to be taken by the control room operator, such as sounding fire alarms, and notifying the shift manager of the type, size, and location of the fire.

c. Control actions to be taken by the fire brigade after notification of a fire, including location to assemble, directions given by the fire brigade leader, the responsibilities of brigade members, such as selection of firefighting and protective equipment, and use of preplanned strategies for fighting fires in specific areas.

d. Control actions to be taken by the security force upon notification of a fire.

e. Define the strategies established for fighting fires in safety-related areas and areas presenting a hazard to safety-related equipment, including the designation of the:

    1. Fire hazards in each plant area/zone covered by a firefighting procedure (prefire plan). Prefire plans use the guidance of NFPA 1620 (Reference 205).

    2. Fire extinguishers best suited for controlling fires with the combustible loadings of each zone and the nearest location of these extinguishers.

    3. Most favorable direction from which to attack a fire in each area in view of the ventilation direction, access hallways, stairs, and doors that are most likely to be free of fire, and the best station or elevation for fighting the fire. Access and egress routes that involve locked doors are specifically identified in the procedure with the appropriate precautions and methods for access specified.

    4. Plant systems that should be managed to reduce the damage potential during a local fire and the location of local and remote controls for such management (e.g., any hydraulic or electrical system in the zone covered by the specific firefighting procedure that could increase the hazards in the area because of overpressurization or electrical hazards).

    5. Vital heat-sensitive system components that need to be kept cool while fighting a local fire. Particularly hazardous combustibles that need cooling are designated.

    6. Potential radiological and toxic hazards in fire zones.

    7. Ventilation system operation that provides desired plant air distribution when the ventilation flow is modified for fire containment or smoke clearing operations.

    8. Operations requiring control room and shift manager coordination or authorization.

9. Instructions for plant operators and other plant personnel during a fire.

f. Organize the fire brigade and assign special duties according to job title so that the firefighting functions are covered for each shift by personnel trained and qualified to perform these functions. These duties include command control of the brigade, transporting fire suppression, and support equipment to the fire scenes, applying the extinguishing agent to the fire, communication with the control room, and coordination with offsite fire departments.

9.5.1.8.4 Control of Combustible Materials, Hazardous Materials, and Ignition Sources

The control of combustible materials is defined by administrative procedures. These procedures impose the following controls:

a. Prohibit the storage of combustible materials (including unused ion exchange resins) in areas that contain or expose safety-related equipment.

b. Govern the handling of and limit transient fire loads such as flammable liquids, wood, and plastic materials in buildings containing safety-related systems or equipment.

c. Assign responsibility to the appropriate supervisor for reviewing work activities to identify transient fire loads.

d. Govern the use of ignition sources by use of a flame permit system to control welding, flame cutting, grinding, brazing and soldering operations, and temporary electrical power cables. A separate permit is issued for each area where such work is done. If work continues over more than one shift, the permit is valid for not more than 24 hours when the plant is operating or for the duration of a particular job during plant shutdown. NFPA 51B (Reference 202) and 241 (Reference 203) are used as guidance.

e. Minimize waste, debris, scrap, and oil spills or other combustibles resulting from a work activity in the safety-related area while work is in progress, and remove the same upon completion of the activity or at the end of each work shift.

f. Govern periodic inspections for accumulation of combustibles for continued compliance with these administrative controls.

g. Prohibit the storage of acetylene-oxygen and other compressed gasses in areas that contain or expose safety-related equipment or the fire protection system that serves those areas. A permit system is required to control the use of this equipment in safety-related areas of the plant.

h. Govern the use and storage of hazardous chemicals in areas that contain or expose safety-related equipment.

i. Control the use of specific combustibles in safety-related areas. Wood used in safety-related areas during maintenance, modification, or refueling operation (such as lay-down blocks or scaffolding) is treated with a flame retardant in accordance with NFPA 703 (Reference 207). Use of wood inside buildings containing systems or equipment important to safety is only permitted when suitable noncombustible substitutes are not available. Equipment or supplies (such as new fuel) shipped in untreated combustible packing containers are unpacked in safety-related areas if required for valid operating reasons. However, combustible materials are removed from the area immediately following unpacking. Such transient combustible material, unless stored in approved containers, is not left unattended during lunch breaks, shift changes, or other similar periods. Loose combustible packing material, such as wood or paper excelsior, or polyethylene sheeting, is placed in metal containers with tight-fitting self-closing metal covers. Only noncombustible panels or flame-retardant tarpaulins or approved materials of equivalent fire-retardant characteristics are used. Any other fabrics or plastic films used are certified to conform to the large-scale fire test described in NFPA 701 (Reference 206).

j. Govern the control of electrical appliances in areas that contain or expose safety-related equipment.

9.5.1.8.5 Control of Radioactive Materials

The plant is designed with provisions for sampling of liquids resulting from fire emergencies that may contain radioactivity and may be released to the environment. Plant operating procedures require such liquids to be collected, sampled, and analyzed prior to discharge. Liquid discharges are required to be below activity limits prior to discharge.

9.5.1.8.6 Testing and Inspection

Testing and inspection requirements are imposed through administrative procedures. Maintenance or modifications to the fire protection system are subject to inspection for conformation to design requirements. 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 (DCD Reference 9.5.5.2) and NFPA 25 (Reference 212). Installation of portions of the system where performance cannot be verified through preoperational tests, such as penetration seals, fire-retardant coatings, cable routing, and fire barriers are inspected. Inspections are performed by individuals knowledgeable of fire protection design and installation requirements. Open flame or combustion-generated smoke is not used for leak testing or similar procedures such as air flow determination. Inspection and testing procedures address the identification of items to be tested or inspected, responsible organizations for the activity, acceptance criteria, documentation requirements and sign-off requirements.

Fire protection materials subject to degradation (such as fire stops, seals, and fire retardant coatings are visually inspected periodically for degradation or damage. Fire hoses are hydrostatically tested in accordance with NFPA 1962 (Reference 201). Hoses stored in outside hose stations are tested annually and interior standpipe hoses are tested every three years.

The fire protection system is periodically tested in accordance with plant procedures. Testing includes periodic operational tests and visual verification of damper and valve positions. Fire doors and their closing and latching mechanisms are also included in these procedures.

STD COL 9.5-6 The preoperational testing program describes the procedures for confirming that the as-installed configuration of fire barriers matches the tested configurations. The procedures describe the process for identifying and dispositioning deviations.

9.5.1.8.7 Personnel Qualification and Training

STD COL 9.5-1 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 Subsection 13.1.1.2.10.

Qualification and training of other plant personnel involved in the fire protection program is governed by plant qualification procedures and is conducted by personnel qualified by training and experience in these areas. These classifications include training personnel, maintenance personnel assigned to work on the fire protection system, and operations personnel assigned to system operation and testing.

9.5.1.8.8 Fire Doors

STD COL 9.5-3 Fire doors separating safety-related areas are self-closing or provided with closing mechanisms and are inspected semiannually to verify that the automatic hold open, release and closing mechanisms and latches are operable. Watertight and missile resistant doors are not provided with closing mechanisms. Fire doors with automatic hold open and release mechanisms are inspected daily to verify that the doorways are free of obstructions.

Fire doors separating safety-related areas are normally closed and latched. Fire doors that are locked closed are inspected weekly to verify position. Fire doors that are closed and latched are inspected daily to ensure that they are in the closed position. Fire doors that are closed and electrically supervised at a continuously manned location are not inspected.

9.5.1.8.9 Emergency Planning

Emergency planning is described in Section 13.3.

VEGP DEP 1.1-1 9.5.1.9 Combined License Information

9.5.1.9.1 Qualification Requirements for Fire Protection Program

STD COL 9.5-1 This COL Item is addressed as follows:

Qualification requirements for individuals responsible for development of the Fire Protection Program are discussed in Subsections 9.5.1.6 and 9.5.1.8.7.

Training of firefighting personnel is discussed in Subsections 9.5.1.8, 9.5.1.8.2, and 9.5.1.8.7.

Administrative procedures and controls governing the Fire Protection Program during plant operation are discussed in Subsections 9.5.1.8.1.2, 9.5.1.8.3, 9.5.1.8.4, 9.5.1.8.5, and 9.5.1.8.6.

Fire protection system maintenance is discussed in Subsection 9.5.1.8.6.

9.5.1.9.2 Fire Protection Analysis Information

VEGP COL 9.5-2 This COL Item is addressed in Subsection 9A.3.3.

9.5.1.9.3 Regulatory Conformance

STD COL 9.5-3 This COL Item is addressed in Subsections 9.5.1.8.1.1, 9.5.1.8.8, and 9.5.1.8.9 and in Table 9.5-201.

9.5.1.9.4 NFPA Exceptions

STD COL 9.5-4 This COL item is addressed in Subsection 9.5.1.8.1.1.

9.5.1.9.6 Verification of Field Installed Fire Barriers

STD COL 9.5-6 This COL Item is addressed in Subsection 9.5.1.8.6.

9.5.1.9.7 Establishment of Procedures to Minimize Risk for Fire Areas Breached During Maintenance

STD COL 9.5-8 This COL item is addressed in Subsection 9.5.1.8.1.2.

Add the following new subsection after DCD Subsection 9.5.2.2.4:

9.5.2.2.5 Offsite Interfaces

VEGP COL 9.5-9
VEGP COL 18.2-2 The Emergency Notification System (ENS) is part of the Federal Telecommunication System (FTS). The FTS does not rely on station switches at VEGP, and therefore the requirements of Bulletin 80-15 are satisfied.

Back-up power for the ENS is provided by the FTS 2001 supplier. On-site systems supporting the FTS system are provided with multiple power sources including diesel and battery backup. ENS phones are located in the Control Room, TSC, and EOF. Dedicated telephone communication links provided by the FTS, and their locations, include:

- NRC Emergency Notification System (ENS)
  - Control Room
  - TSC
  - EOF
- NRC Health Physics Network (HPN)
  - TSC
  - EOF
- Reactor Safety Counterpart Link (RSCL)
  - TSC

- EOF

- Protective Measures Counterpart Link (PMCL)
  - TSC
  - EOF
- Management Counterpart Link (MCL)
  - TSC
  - EOF
- Operations Center LAN (OCL)
  - TSC
  - EOF

VEGP COL 9.5-10 Design specifications include provisions for multiple power sources for the communication system. The design provides for back-up power to be provided by a combination of diesel generator and/or battery supplied power. Communication system power supplies will be identified in Emergency Implementing Procedures.

The Emergency Notification Network (ENN) is the primary means for communication between the Site and the local authorities, including the State of South Carolina and Aiken, Barnwell, and Allendale Counties, and the State of Georgia and Burke County. The ENN system is available on a twenty-four seven basis. Commercial telephones provide backup for the dedicated telephone circuits.

In the event of failure of either the dedicated telephone circuits or the ENN equipment (transmitting station and/or single/multiple receiving stations), commercial telephone lines are used. To transfer to the back-up system, the communicator manually initiates communications using commercial telephone lines for the affected station(s) as directed by Emergency Implementing Procedures.

9.5.2.5 Combined License Information

9.5.2.5.1 Offsite Interfaces

VEGP COL 9.5-9 This COL Item is addressed in ESPA Emergency Plan Section F and FSAR Subsection 9.5.2.2.5.

9.5.2.5.2 Emergency Offsite Communications

VEGP COL 9.5-10 This COL Item is addressed in ESPA Emergency Plan Section F and FSAR Subsection 9.5.2.2.5.

9.5.2.5.3 Security Communications

STD COL 9.5-11 This COL Item is addressed in the Physical Security Plan.

Add the following subsection after DCD Subsection 9.5.4.5.1.

9.5.4.5.2 Fuel Oil Quality

STD COL 9.5-13 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 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 D4176. In addition, kinematic 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.

9.5.4.7 Combined License Information

9.5.4.7.2 Fuel Degradation Protection

STD COL 9.5-13 This COL Item is addressed in Subsection 9.5.4.5.2.

9.5.5 REFERENCES

201. National Fire Protection Association, “Standard for Inspection, Care, and Use of Fire Hose, Couplings, and Nozzles and the Service Testing of Fire Hose,” NFPA 1962, 2003.

202. National Fire Protection Association, “Standard for Fire Prevention During Welding, Cutting, and Other Hot Work,” NFPA 51B, 2003.

203. National Fire Protection Association, “Standard for Safeguarding Construction, Alteration, and Demolition Operations,” NFPA 241, 2004.

204. National Fire Protection Association, “Standard on Industrial Fire Brigades,” NFPA 600, 2005.

205. National Fire Protection Association, “Recommended Practice for Pre-incident Planning,” NFPA 1620, 2003.

206. National Fire Protection Association, “Standard Methods of Fire Tests for Flame Propagation of Textiles and Films,” NFPA 701, 2004.

207. National Fire Protection Association, “Standard for Fire-Retardant Treated Wood and Fire-Retardant Coatings for Building Materials,” NFPA 703, 2006.

208. National Fire Protection Association, “Standard for Fire Service Respiratory Protection Training,” NFPA 1404, 2006.

209. National Fire Protection Association, “Standard on Training for Initial Emergency Scene Operations,” NFPA 1410, 2005.

210. National Fire Protection Association, “Standard on Fire Department Occupational Safety and Health Program,” NFPA 1500, 2007.

211. National Fire Protection Association, “Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants,” NFPA 804, 2001.

212. National Fire Protection Association, “Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems,” NFPA 25, 2008.

STD COL 9.5-3
STD COL 9.5-4

Table 9.5-201[a] (Sheet 1 of 7)
AP1000 Fire Protection Program Compliance with
BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline	Paragraph	Comp	Remarks
Fire Protection Program			
1. Direction of fire protection program; availability of personnel.	C.1.a(1)	C	Comply. Subsections 9.5.1.8.1.2 and 13.1.1.2.10 address this requirement.
2. Defense-in-depth concept; objective of fire protection program.	C.1.a(2)	C	Comply. Subsections 9.5.1.8 and 9.5.1.8.1 address this requirement.
3. Management responsibility for overall fire protection program; delegation of responsibility to staff.	C.1.a(3)	C	Comply. Subsections 9.5.1.8.1.2 and 13.1.1.2.10
4. The staff should be responsible for: a. Fire protection program requirements. b. Post-fire shutdown capability. c. Design, maintenance, surveillance, and quality assurance of fire protection features. d. Fire prevention activities. e. Fire brigade organization and training. f. Prefire planning.	C.1.a(3)	C	Comply. Subsection 13.1.1.2.10 addresses this requirement.

STD COL 9.5-3

STD COL 9.5-4

Table 9.5-201[a] (Sheet 2 of 7)
AP1000 Fire Protection Program Compliance with
BTP CMEB 9.5-1

	BTP CMEB 9.5-1 Guideline	Paragraph	Comp	Remarks
5.	The organizational responsibilities and lines of communication pertaining to fire protection should be defined through the use of organizational charts and functional descriptions.	C.1.a(4)	C	Comply. Organization and lines of communication are addressed in Figure 13.1-201. Functional descriptions are addressed in Subsections 13.1.1.2.10, 13.1.1.3.1.3, and 13.1.2.1.5.
6.	Personnel qualification requirements for fire protection engineer, reporting to the position responsible for formulation and implementation of the fire protection program.	C.1.a(5)(a)	C	Comply. Subsection 13.1.1.2.10 addresses this requirement.
7.	The fire brigade members' qualifications should include a physical examination for performing strenuous activity, and the training described in Position C.3.d.	C.1.a(5)(b)	C	Comply. Subsections 9.5.1.8.2.1 and 9.5.1.8.2.2 addresses this requirement.
8.	The personnel responsible for the maintenance and testing of the fire protection systems should be qualified by training and experience for such work.	C.1.a(5)(c)	C	Comply. Subsection 9.5.1.8.7 addresses this requirement.
9.	The personnel responsible for the training of the fire brigade should be qualified by training and experience for such work.	C.1.a(5)(d)	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
10.	The following NFPA publications should be used for guidance to develop the fire protection program: No. 4, No. 4A, No. 6, No. 7, No. 8, and No. 27.	C.1.a(6)	C	Alternate Compliance. The NFPA codes cited in BTP CMEB 9.5-1 are historical. Current NFPA codes are referenced for guidance for the fire protection program. Subsection 9.5.1.8.1.1 addresses this requirement.

STD COL 9.5-3

STD COL 9.5-4

Table 9.5-201[(a)] (Sheet 3 of 7)
AP1000 Fire Protection Program Compliance with
BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline		Paragraph	Comp	Remarks
11.	On sites where there is an operating reactor, and construction or modification of other units is underway, the superintendent of the operating plant should have a lead responsibility for site fire protection.	C.1.a(7)	C	Comply. Subsection 13.1.1.2.10 addresses this requirement.
Fire Protection Analysis				
14.	Fires involving facilities shared between units should be considered.	C.1.b	C	Comply. The FHA demonstrates the plant's ability to perform safe shutdown functions and minimize radioactive releases to the environment. Postulated fires in shared facilities that do not contain SSCs important to safety and do not contain radioactive materials do not affect these functions.
15.	Fires due to man-made site-related events that have a reasonable probability of occurring and affecting more than one reactor unit should be considered.	C.1.b	C	Comply. Subsections 2.2 and 3.5 establish that these events are not credible.
Fire Suppression System Design Basis				
22.	Fire protection systems should retain their original design capability for potential man-made, site-related events that have a reasonable probability of occurring at a specific plant site.	C.1.c(4)	C	Comply. Subsections 2.2 and 3.5 establish that these events are not credible.
Fire Protection Program Implementation				
26.	The fire protection program for buildings storing new reactor fuel and for adjacent fire areas that could affect the fuel storage area should be fully operational before fuel is received at the site.	C.1.e(1)	C	Comply. Subsection 9.5.1.8.1 addresses this requirement.

STD COL 9.5-3

STD COL 9.5-4

Table 9.5-201[(a)] (Sheet 4 of 7)
AP1000 Fire Protection Program Compliance with
BTP CMEB 9.5-1

	BTP CMEB 9.5-1 Guideline	Paragraph	Comp	Remarks
27.	The fire protection program for an entire reactor unit should be fully operational prior to initial fuel loading in that unit.	C.1.e(2)	C	Comply. Subsection 9.5.1.8.1 addresses this requirement.
28.	Special considerations for the fire protection program on reactor sites where there is an operating reactor and construction or modification of other units is under way.	C.1.e(3)	C	Comply. Subsection 9.5.1.8.1.2. m addresses this requirement.
29.	Establishing administrative controls to maintain the performance of the fire protection system and personnel.	C.2	C	Comply. Subsection 9.5.1.8.1.2 addresses this requirement.
Fire Brigade				
30.	The guidance in Regulatory Guide 1.101 should be followed as applicable.	C.3.a	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
31.	Establishing site brigade: minimum number of fire brigade members on each shift; qualification of fire brigade members; competence of brigade leader.	C.3.b	C	Comply. Subsection 9.5.1.8.2.2 and 13.1.2.1.5 address this requirement.
32.	The minimum equipment provided for the brigade should consist of turnout coats, boots, gloves, hard hats, emergency communications equipment, portable ventilation equipment, and portable extinguishers.	C.3.c	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
33.	Recommendations for breathing apparatus for fire brigade, damage control, and control room personnel.	C.3.c	C	Comply. Subsection 9.5.1.8.2.2 and DCD Subsections 6.4.2.3 and 6.4.4 address these requirements.

STD COL 9.5-3

STD COL 9.5-4

Table 9.5-201[a] (Sheet 5 of 7)
AP1000 Fire Protection Program Compliance with
BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline		Paragraph	Comp	Remarks
34.	Recommendations for the fire brigade training program.	C.3.d	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
Quality Assurance Program				
35.	Establishing quality assurance (QA) programs by applicants and contractors for the fire protection systems for safety-related areas; identification of specific criteria for quality assurance programs.	C.4	C	Comply. DCD Subsection 9.5.1.7 and Chapter 17 address this requirement.
Building Design				
50.	Fire doors should be inspected semiannually to verify that automatic hold-open, release, and closing mechanisms and latches are operable.	C.5.a (5)	C	Comply. Subsection 9.5.1.8.8 addresses this requirement.
51.	Alternative means for verifying that fire doors protect the door opening as required in case of fire.	C.5.a (5)	C	Comply. Subsection 9.5.1.8.8 addresses this requirement.
52.	The fire brigade leader should have ready access to keys for any locked fire doors.	C.5.a (5)	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
55.	Stairwells serving as escape routes, access routes for firefighting, or access routes to areas containing equipment necessary for safe shutdown should be enclosed in masonry or concrete towers with a minimum fire resistance rating of 2 hours and self-closing Class B fire doors.	C.5.A (6)	C	Comply. Subsection 9A.3.3 addresses this requirement for miscellaneous buildings located in the yard.
56.	Fire exit routes should be clearly marked.	C.5.a (7)	C	Comply. DCD Subsection 9.5.1.2.1.1 addresses this requirement.

STD COL 9.5-3

STD COL 9.5-4

Table 9.5-201[(a)] (Sheet 6 of 7)
AP1000 Fire Protection Program Compliance with
BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline		Paragraph	Comp	Remarks
71.	Water drainage from areas that may contain radioactivity should be collected, sampled, and analyzed before discharge to the environment.	C.5.a(14)	C	Comply. Capability is provided. Subsection 9.5.1.8.5 addresses this requirement.
Control of Combustibles				
80.	Use of compressed gases inside buildings should be controlled.	C.5.d (2)	C	Comply. Subsection 9.5.1.8.4.g addresses this requirement.
Lighting and Communication				
111.	A portable radio communications system should be provided for use by the fire brigade and other operations personnel required to achieve safe plant shutdown.	C.5.g (4)	C	Comply. Subsections 9.5.1.8.1.2.a.3.v, 9.5.1.8.2.2, and DCD Subsections 9.5.2 and 9.5.2.2.1 address this requirement.
Water Sprinkler and Hose Standpipe Systems				
149.	All valves in the fire protection system should be periodically checked to verify position.	C.6.c (2)	C	Comply. Subsection 9.5.1.8.6 addresses this requirement.
157.	The fire hose should be hydrostatically tested in accordance with NFPA 1962. Hoses stored in outside hose houses should be tested annually. The interior standpipe hose should be tested every 3 years.	C.6.c (6)	C	Comply. Subsection 9.5.1.8.6 addresses this requirement.
Primary and Secondary Containment				
174.	Self-contained breathing apparatus should be provided near the containment entrances for fire fighting and damage control personnel. These units should be independent of any breathing apparatus provided for general plant activities.	C.7.a (2)	C	Comply. Subsection 9.5.1.8.2.2 addresses this requirement.
Main Control Room Complex				

STD COL 9.5-3

STD COL 9.5-4

Table 9.5-201[(a)] (Sheet 7 of 7)
AP1000 Fire Protection Program Compliance with
BTP CMEB 9.5-1

BTP CMEB 9.5-1 Guideline		Paragraph	Comp	Remarks
180.	Breathing apparatus for main control room operators should be readily available.	C.7.b	C	Comply. DCD Subsection 6.4.2.3 addresses this requirement.
Cooling Towers				
225.	Cooling towers should be of noncombustible construction or so located and protected that a fire will not adversely affect any safety-related systems or equipment.	C.7.q	C	Comply. Subsection 9A.3.3 addresses this requirement.
Storage of Acetylene-Oxygen Fuel Gases				
228.	Gas cylinder storage locations should not be in areas that contain or expose safety-related equipment or the fire protection systems that serve those safety-related areas.	C.8.a	C	Comply. Subsection 9.5.1.8.4.g addresses this requirement.
229.	A permit system should be required to use this equipment in safety-related areas of the plant.	C.8.a	C	Comply. Subsection 9.5.1.8.4.g addresses this requirement.
Storage Areas for Ion Exchange Resins				
230.	Unused ion exchange resins should not be stored in areas that contain or expose safety-related equipment.	C.8.b	C	Comply. Subsection 9.5.1.8.4.a addresses this requirement.
Hazardous Chemicals				
231.	Hazardous chemicals should not be stored in areas that contain or expose safety-related equipment.	C.8.c	C	Comply. Subsection 9.5.1.8.4.h addresses this requirement.

a) This table supplements DCD Table 9.5.1-1.

STD COL 9.5-4

Table 9.5-202[a]
Exceptions to NFPA Standard Requirements

Requirement	AP1000 Exception or Clarification
NFPA 804 (Reference 211) contains requirements specific to light water reactors.	Compliance with portions of this standard is as identified within DCD Section 9.5.1 and WCAP-15871. The intake structure is non-combustible construction, does not provide any safety function, and does not contain any equipment important to safety. Automatic sprinkler protection is not warranted and is not provided.

a) This table supplements DCD Table 9.5.1-3.

APPENDIX 9A
FIRE PROTECTION ANALYSIS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

9A.2 FIRE PROTECTION METHODOLOGY

9A.2.1 Fire Area Description

Add the following information at the end of the first paragraph in DCD Subsection 9A.2.1:

VEGP DEP 18.8-1 Figure 9A-201 replaces DCD Figure 9A-3 (Sheet 1), to reflect the relocation of the Operations Support Center.

9A.3.3 Yard Area and Outlying Buildings

Replace the second sentence of Subsection 9A.3.3 with the following information.

VEGP COL 9.5-2 Miscellaneous yard areas do not contain safety-related components or systems, do not contain radioactive materials, and are located such that a fire or effects of a fire, including smoke, do not adversely affect any safety-related systems or equipment. Miscellaneous areas include such structures, for example, as maintenance shops, warehouses, the administrative building, training/office centers, and flammable and combustible material storage tanks. The intake structure is nonsafety-related, does not contain any safety-related equipment, and is remotely located from safety-related structures, systems and components. The miscellaneous areas are located outside of the nuclear island, which is separated from the other yard areas by 3-hour fire rated barriers. Fire detection and suppression are provided as determined by the fire hazards analysis and applicable building codes and insurance company loss prevention standards. Water-based fire suppression systems are supplied by a separate yard main that is independent of AP1000 fire loops. The separate yard main is designed and constructed in accordance with applicable NFPA and local codes and applicable insurance company loss prevention standards.

The cooling tower is not used as the ultimate heat sink or for fire protection purposes. Therefore, the guidance specified in BTP CMEB 9.5-1 is not applicable. The cooling tower serves no safety function and has no safety design basis. The cooling tower does not contain any equipment capable of releasing radioactivity to the atmosphere. The cooling tower fill is a PVC material with a flame spread rating of 25 or less. The cooling tower is remotely located from HVAC air intakes such that smoke and products of combustion do not affect any safety-related plant areas.

STD COL 9.5-3 Stairwells in miscellaneous buildings located in the yard serving as escape routes or access routes for firefighting, are enclosed in masonry or concrete towers with a minimum fire resistance rating of 2 hours and self-closing Class B fire doors. The two hour fire-resistance rating for the masonry or concrete material is based on testing conducted in accordance with ASTM E119 (Reference 201) and NFPA 251 (Reference 202).

9A.4 REFERENCES

201. American Society of Mechanical Engineers, “Standard Test Methods for Fire Tests of Building Construction and Materials,” ASTM E119-08a.

202. National Fire Protection Association, “Standard Methods of Tests of Fire Endurance of Building Construction and Materials,” NFPA 251, 2006.

Security-Related Information — Withheld Under 10 CFR 2.390(d)
(See Part 9 of this COL Application)

(Note: This figure replaces DCD Figure 9A-3 Sheet 1 of 3.)

VEGP DEP 18.8-1

Figure 9A-201
[*Annex I & II Building Fire Areas Plan at Elevation 100'-0" & 107'-2"*]*

***NRC staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.**

CHAPTER 10

STEAM AND POWER CONVERSION

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CHAPTER 10
STEAM AND POWER CONVERSION

10.1 SUMMARY DESCRIPTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

10.1.3 COMBINED LICENSE INFORMATION ON EROSION-CORROSION MONITORING

Add the following text at the end of DCD Subsection 10.1.3.

10.1.3.1 Erosion-Corrosion Monitoring

STD COL 10.1-1 The flow accelerated corrosion (FAC) monitoring program analyzes, inspects, monitors and trends those nuclear power plant components that are potentially susceptible to erosion-corrosion damage such as carbon steel components that carry wet steam. 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 (Reference 201). At a minimum, a CHECWORKS type Pass 1 analysis is used for low and highly susceptible FAC locations and a CHECWORKS type Pass 2 analysis is used for highly susceptible FAC locations when Pass 1 analysis results warrant. 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.

10.1.3.1.1 Analysis

An industry-sponsored program is used to identify the most susceptible components and to evaluate the rate of wall thinning for components and piping potentially susceptible to FAC. Each susceptible component is tracked in a database and is inspected, based on susceptibility. Analytical methods utilize the results of plant-specific inspection data to develop plant-specific correction factors. This correction accounts for uncertainties in plant data, and for systematic discrepancies caused by plant operation. For each piping component, the analytical method predicts the wear rate, and the estimated time until it must be re-inspected, repaired, or replaced. Carbon steel piping (ASME III and B31.1) that is used for single or multi-phase high temperature flow are the most susceptible to erosion-corrosion damage and receive the most critical analysis.

10.1.3.1.2 Industry Experience

Review and incorporation of industry experience provides a valuable supplement to plant analysis. Industry experience is used to update the program by identifying susceptible components or piping features.

10.1.3.1.3 Inspections

Wall thickness measurements establish the extent of wear in a given component, provide data to help evaluate trends, and provide data to refine the predictive model. Components are inspected for wear using ultrasonic techniques (UT), radiography techniques (RT), or by visual observation. The initial inspections are used as a baseline for later inspections. Each subsequent inspection determines the wear rate for the piping and components and the need for inspection frequency adjustment for those components.

10.1.3.1.4 Training and Engineering Judgement

The FAC program is administered by both trained and experienced personnel. Task specific training is provided for plant personnel that implement the monitoring program. Specific non-destructive examination (NDE) is carried out by personnel qualified in the given NDE method. Inspection data is analyzed by engineers or other experienced personnel to determine the overall effect on the system or component.

10.1.3.1.5 Long-Term Strategy

This strategy focuses on reducing wear rates and performing inspections on the most susceptible locations.

10.1.3.2 Procedures

10.1.3.2.1 Generic Plant Procedure

The FAC monitoring program is governed by procedure. This procedure contains the following elements:

- A requirement to monitor and control FAC.
- Identification of the tasks to be performed and associated responsibilities.
- Identification of the position that has overall responsibility for the FAC monitoring program at each plant.

- Communication requirements between the coordinator and other departments that have responsibility for performing support tasks.
- Quality Assurance requirements.
- Identification of long-term goals and strategies for reducing high FAC wear rates.
- A method for evaluating plant performance against long-term goals.

10.1.3.2.2 Implementing Procedures

The FAC implementing procedures provide guidelines for controlling the major tasks. The plant procedures for major tasks are as follows:

- Identifying susceptible systems.
- Performing FAC analysis.
- Selecting and scheduling components for initial inspection.
- Performing inspections.
- Evaluating degraded components.
- Repairing and replacing components when necessary.
- Selecting and scheduling locations for the follow-on inspections.
- Collection and storage of inspections records.

10.1.3.3 Plant Chemistry

The responsibility for system chemistry is under the purview of the plant chemistry section. The plant chemistry section specifies chemical addition in accordance with plant procedures.

Add the following after DCD Subsection 10.1.3:

10.1.4 REFERENCES

201. EPRI NSAC-202L-R3, Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R3), Electric Power Research Institute (EPRI) Technical Report 1011838, Palo Alto, CA, 2006.

10.2 TURBINE-GENERATOR

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

10.2.2 SYSTEM DESCRIPTION

Add the following sentence at the end of the second paragraph of DCD Subsection 10.2.2.

STD SUP 10.2-1 Subsection 3.5.1.3 addresses the probability of generation of a turbine missile for AP1000 plants in a side-by-side configuration.

Add the following statement at the end of DCD Subsection 10.2.2.

STD SUP 10.2-4 Preoperational and startup tests provide guidance to operations personnel to ensure the proper operability of the turbine generator system.

10.2.3 TURBINE ROTOR INTEGRITY

Add the following statement at the end of DCD Subsection 10.2.3.

STD SUP 10.2-5 Operations and maintenance procedures mitigate the following potential degradation mechanisms in the turbine rotor and buckets/blades: pitting, stress corrosion cracking, corrosion fatigue, low-cycle fatigue, erosion, and erosion-corrosion.

10.2.3.6 Maintenance and Inspection Program Plan

Add the following at the end of DCD Subsection 10.2.3.6.

STD SUP 10.2-3 The inservice inspection (ISI) program for the turbine assembly provides assurance that rotor flaws that lead to brittle fracture of a rotor are detected. The ISI program also coincides with the ISI schedule during shutdown, as required by the ASME Boiler and Pressure Vessel Code, Section XI, and includes complete inspection of all significant turbine components, such as couplings, coupling bolts, turbine shafts, low-pressure turbine blades, low-pressure rotors, and high-pressure rotors. This inspection consists of visual, surface, and volumetric examinations required by the code.

10.2.6 COMBINED LICENSE INFORMATION ON TURBINE MAINTENANCE AND INSPECTION

Replace the text in DCD Subsection 10.2.6 with the following:

STD COL 10.2-1 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.

10.3 MAIN STEAM SUPPLY SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

10.3.2.2.1 Main Steam Piping

Add the following at the end of DCD Subsection 10.3.2.2.1.

STD SUP 10.3-1 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

10.3.5.4 Chemical Addition

Add the following at the end of DCD Subsection 10.3.5.4.

STD SUP 10.3-2 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.

10.3.6.2 Material Selection and Fabrication

Add the following at the end of DCD Subsection 10.3.6.2.

STD SUP 10.3-3 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 intergranular stress-corrosion cracking by controlling chemicals that are used on system components.

10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

10.4.2.2.1 General Description

Revise the first sentence of the third paragraph of DCD Subsection 10.4.2.2.1 to remove the brackets.

VEGP CDI The circulating water system (CWS) provides the cooling water for the vacuum pump seal water heat exchangers.

10.4.2.2.2 Component Description

Revise the fourth sentence of the first paragraph of DCD Subsection 10.4.2.2.2 to remove the brackets.

VEGP CDI Seal water flows through the shell side of the seal water heat exchanger and circulating water flows through the tube side.

Subsection 10.4.5 is modified using full text incorporation to provide site specific information to replace the DCD conceptual design information (CDI).

DCD

10.4.5 CIRCULATING WATER SYSTEM

10.4.5.1 Design Basis

10.4.5.1.1 Safety Design Basis

The circulating water system (CWS) serves no safety-related function and therefore has no nuclear safety design basis.

10.4.5.1.2 Power Generation Design Basis

VEGP CDI

The CWS supplies cooling water to remove heat from the main condensers. The CWS also supplies cooling water to 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.

DCD

10.4.5.2 System Description

10.4.5.2.1 General Description

Classification of components and equipment in the circulating water system is given in Section 3.2.

VEGP COL 10.4-1

The CWS and the cooling tower provide a heat sink for the waste heat exhausted from the steam turbine. Additional cooling is supplied from the CWS through a tap in the main supply header for the TCS heat exchangers and the condenser vacuum pump seal water heat exchangers. CWS design parameters are provided in Table 10.4-201 and Table 10.4-202.

VEGP CDI

The CWS consists of three 33-1/3-percent-capacity circulating water pumps, one hyperbolic natural draft cooling tower (NDCT), and associated piping, valves, and instrumentation.

DCD

Makeup water to the CWS is provided by the raw water system (RWS). In addition, water chemistry is controlled by a local chemical feed system.

10.4.5.2.2 Component Description

Circulating Water Pumps

VEGP CDI

The three circulating water pumps are vertical, wet pit, single-stage, mixed-flow pumps driven by electric motors. The pumps are mounted in a pump pit, which is attached to the cooling tower basin. The three pump discharge lines connect to a common header which connects to the two inlet water boxes of the condenser and may also supply cooling water to the TCS and condenser vacuum pump seal water heat exchangers. Each pump discharge line has a motor-operated butterfly valve located between the pump discharge and the main header. This permits isolation of one pump for maintenance and allows for two-pump operation.

Cooling Tower

The hyperbolic NDCT is designed to reject the full-load waste heat to the atmosphere by evaporation as the circulating water passes through the heat exchanger section. The cooling tower is designed to cool the circulating water to 91°F based on a design wet bulb temperature of 80°F.

The cooling tower is located approximately 1000 ft. south of the plant and has a basin water level of approximately 219 ft. MSL. Should a cooling tower basin wall break, little, if any, water would reach the plant because of the remote location of the tower and the grading of the site. The height of the cooling tower is approximately 600 ft., thus, there is no potential for the cooling tower to fall and damage safety related structures or components. Because of the remote location, the cooling tower height, and the buoyant rise of the plumes, the plumes will dissipate before they interfere with the SWS cooling towers intake, any plant ventilation intake, or the plant switchyard. Because of the height of cooling tower it is unlikely there will be fogging near the plant as a result of the cooling tower plume.

Cooling Tower Makeup and Blowdown

DCD The circulating water system makeup is provided by the raw water system.

VEGP CDI Makeup to and blowdown from the CWS is controlled by the makeup and blowdown control valves. The makeup control valves maintain the cooling tower basin level. These valves, along with a local chemical feed system, provide chemistry control in the circulating water in order to maintain a noncorrosive, nonscale-forming condition and limit biological growth in CWS components.

DCD **Piping and Valves**

VEGP CDI The underground portions of the CWS piping are constructed of prestressed concrete pressure piping. The remainder of the piping is carbon steel and is coated internally with a corrosion-resistant compound.

VEGP COL 10.4-1 The primary drainage path for the condenser water boxes and tube bundles is via gravity to the cooling tower basins using the cooling tower bypass lines. As an alternate, condenser water box drain lines are provided that direct drainage to the turbine building sumps.

DCD Motor-operated butterfly valves are provided in each of the circulating water lines at their inlet to and exit from the condenser shell to allow isolation of portions of the condenser.

VEGP CDI Control valves provide regulation of cooling tower makeup and blowdown.

DCD The circulating water system is designed to withstand the maximum operating discharge pressure of the circulating water pumps.

VEGP CDI Piping includes the expansion joints, butterfly valves, condenser water boxes, and tube bundles.

The piping design pressure of the CWS is 115 psig.

DCD

Circulating Water Chemical Injection

Circulating water chemistry is maintained by a local chemical feed skid at the CWS cooling tower.

VEGP CDI Circulating water system chemical feed equipment injects the required chemicals into the circulating water at the CWS cooling tower basin area.

DCD This maintains a noncorrosive, nonscale-forming condition and limits the biological film formation that reduces the heat transfer rate in the condenser and the heat exchangers supplied by the circulating water system.

VEGP COL 10.4-1 The specific chemicals used within the system are determined by the site water conditions and are monitored by plant chemistry personnel.

DCD The chemicals can be divided into six categories based upon function: biocide, algaecide, pH adjuster, corrosion inhibitor, scale inhibitor, and a silt dispersant. The pH adjuster, corrosion inhibitor, scale inhibitor, and dispersant are metered into the system continuously or as required to maintain proper concentrations. The biocide application frequency may vary with seasons.

VEGP CDI The algaecide is applied, as necessary, to control algae formation on the cooling tower.

VEGP COL 10.4-1 The following chemicals are used to control circulating water chemistry:

- Biocide - Sodium hypochlorite, sodium bromide, stabilized bromine
- Algaecide - Biocide treatment is adequate for algae control
- pH Adjuster - No pH adjustment expected to be required at anticipated cycles of concentration
- Corrosion Inhibitor - Proprietary blend of phosphonate, phosphinosuccinic oligomer (PSO) and phosphoric acid
- Scale Inhibitor - Sodium Hexametaphosphate
- Silt Dispersant - Proprietary tagged high-strength polymer

DCD Addition of biocide and water treatment chemicals is performed by local chemical feed injection metering pumps and is adjusted as required.

VEGP CDI Chemical concentrations are measured through analysis of grab samples from the CWS.

DCD Residual chlorine is measured to monitor the effectiveness of the biocide treatment.

10.4.5.2.3 System Operation

VEGP CDI The three circulating water pumps take suction from the circulating water pump pit and circulate the water through the tube side of the main condenser, with smaller flows to the TCS, the condenser vacuum pump seal water heat exchangers, and back through the piping discharge network to the NDCT. See Figure 10.4-201. The NDCT cools the circulating water by discharging the water through nozzles in the tower distribution headers. The water is then diffused through fill material to the basin beneath the tower and, in the process, rejects heat to the atmosphere. Provisions are made during freezing weather to minimize the possibility of ice accumulation.

A 100 percent capacity bypass system has also been provided on the warm water inlet to prevent icing of the fill during a freezing weather start-up. When the tower is started during freezing weather, the warm water flow will bypass the fill into the cold water basin through the bypass line. The bypass is normally used only during plant startup in freezing weather or to maintain CWS temperature above 40°F while operating at partial load during periods of freezing weather.

VEGP CDI The raw water system supplies makeup water to the cooling tower basin to replace water losses due to evaporation, wind drift, and blowdown. A separate connection is also provided between the RWS and CWS to fill the CWS piping.

DCD A condenser tube cleaning system is installed to clean the circulating water side of the main condenser tubes.

VEGP CDI Blowdown from the CWS is taken from the discharge of the CWS pumps and is discharged to the plant outfall.

DCD The circulating water system is used to supply cooling water to the main condenser to condense the steam exhausted from the main turbine.

VEGP CDI If the circulating water pumps, the cooling tower, or the circulating water piping malfunction such that condenser backpressure rises above the maximum allowable value, the main condenser will no longer be able to adequately support unit operation.

DCD

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 when the condenser is not available.

Passage of condensate from the main condenser into the circulating water system through a condenser tube leak is not possible during power generation operation, since the circulating water system operates at a greater pressure than the condenser.

VEGP CDI

Turbine building closed cooling water in the TCS heat exchangers is maintained at a higher pressure than the circulating water to prevent leakage of the circulating water into the closed cooling water system.

Cooling water to the condenser vacuum pump seal water heat exchangers is supplied from the circulating water system. Cooling water flow from the circulating water system is normally maintained through all four heat exchangers to facilitate placing the spare condenser vacuum pump in service.

DCD

Isolation valves are provided for the condenser vacuum pump seal water heat exchanger cooling water supply lines to facilitate maintenance.

Small circulating water system leaks in the turbine building will drain into the waste water system. Large circulating water system leaks due to pipe failures will be indicated in the control room by a loss of vacuum in the condenser shell. The effects of flooding due to a circulating water system failure, such as the rupture of an expansion joint, will not result in detrimental effects on safety-related equipment since there is no safety-related equipment in the turbine building and the base slab of the turbine building is located at grade elevation. Water from a system rupture will run out of the building through a relief panel in the turbine building west wall before the level could rise high enough to cause damage. Site grading will carry the water away from safety-related buildings.

VEGP CDI

The cooling tower is located so that collapse of the tower has no potential to damage equipment, components, or structures required for safe shutdown of the plant.

DCD

10.4.5.3 Safety Evaluation

The circulating water system has no safety-related function and therefore requires no nuclear safety evaluation.

10.4.5.4 Tests and Inspections

Components of the circulating water system are accessible as required for inspection during plant power generation.

VEGP CDI

The circulating water pumps are tested in accordance with standards of the Hydraulic Institute.

DCD

Performance, hydrostatic, and leakage tests associated with preinstallation and preoperational testing are performed on the circulating water system. The system performance and structural and leaktight integrity of system components are demonstrated by continuous operation.

10.4.5.5 Instrumentation Applications

VEGP CDI

Instrumentation provided indicates the open and closed positions of motor-operated butterfly valves in the circulating water piping. 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.

Local grab samples are used to periodically test the circulating water quality to limit harmful effects to the system piping and valves due to improper water chemistry.

Pressure indication is provided on the circulating water pump discharge lines.

DCD

A differential pressure transmitter is provided between one inlet and outlet branch to the condenser. This differential pressure transmitter is used to determine the frequency of operating the condenser tube cleaning system (CES).

VEGP CDI

Temperature indication is supplied on the common CWS inlet header to the TCS heat exchanger trains. This temperature is also representative of the inlet cooling water temperature to the main condenser.

A flow element is provided on the common discharge line from the TCS heat exchangers to allow monitoring of the total flow through the TCS heat exchangers. Flow measurement for the raw water makeup to the cooling tower and for the cooling tower blowdown is also provided.

VEGP CDI Level instrumentation provided in the circulating water pump intake structure activates makeup flow from the RWS to the cooling tower basin when required. Level instrumentation also annunciates a low-water level in the pump structure and a high-water level in the cooling tower basin.

VEGP COL 10.4-1 The circulating water chemistry is controlled by regulating cooling tower blowdown and by chemical addition, to maintain the circulating water with an acceptable Langelier Index and is maintained in range established by plant chemistry personnel.

VEGP CDI The system accomplishes this by regulating the blowdown valve. This regulation causes the tower basin water level to fluctuate. This fluctuation is sensed by a level controller that operates the makeup valve to the cooling tower basin.

DCD The control approach is to allow the makeup water to concentrate naturally to its upper limit. Provisions are made to add chemicals for pH control.

VEGP CDI The cycles of concentration at which the cooling tower is operated is dependent on the quality of the cooling tower makeup water. Cooling tower blowdown is discharged to the Savannah River via the waste water system.

DCD Monitoring of the circulating water system is performed through the data display and processing system. Control functions are performed by the plant control system. Appropriate alarms and displays are available in the control room. See Chapter 7.

10.4.7.2.1 General Description

Replace the last sentence of the sixth paragraph of DCD Subsection 10.4.7.2.1 as follows.

VEGP COL 10.4-2 The oxygen scavenger agent is hydrazine and the pH control agent is methoxypropylamine (MPA). During shutdown conditions, carbohydrazide may be used in place of hydrazine.

STD SUP 10.4-2 Oxygen scavenging and ammoniating agents are selected and utilized for plant secondary water chemistry optimization following the guidance of NEI-97-06, "Steam Generator Program Guidelines" (Reference 201). The EPRI Pressurized Water Reactor Secondary Water Chemistry Guidelines are followed as described in NEI 97-06.

Add new paragraph at the end of the DCD Subsection 10.4.7.2.1:

STD SUP 10.4-1 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

10.4.12 COMBINED LICENSE INFORMATION

10.4.12.1 Circulating Water System

VEGP COL 10.4-1 This COL Item is addressed in Subsection 10.4.5.2.1, 10.4.5.2.2, and 10.4.5.5.

10.4.12.2 Condensate, Feedwater and Auxiliary Steam System Chemistry Control.

VEGP COL 10.4-2 This COL Item is addressed in Subsection 10.4.7.2.1.

10.4.12.3 Potable Water

Replace the entire paragraph for DCD Subsection 10.4.12.3 with the following.

VEGP COL 10.4-3 Potable water is produced on site by the Potable Water System (PWS). Source water for the PWS is from the well water subsystem to the Raw Water System. Sodium hypochlorite is used as the biocide. The PWS is discussed in Subsection 9.2.5.

10.4.13 REFERENCES

201. Nuclear Energy Institute, "Steam Generator Program Guidelines," NEI 97-06, Revision 2, May 2005.

VEGP CDI

Table 10.4-201
Supplemental Main Condenser Design Data

Condenser Data	
Circulating water flow	600,000 gpm

Note: This table supplements DCD Table 10.4.1-1.

VEGP COL 10.4-1

Table 10.4-202
Supplemental Design Parameters for Major Circulating Water System Components

Circulating Water Pump	
Quantity	Three per unit
Flow rate (gal/min)	215,180 (note 2)
Natural Draft Cooling Tower	
Quantity	One per unit
Approach temperature (°F)	11
Inlet temperature (°F)	114.9 (note 3)
Outlet temperature (°F)	91
Approximate Temperature range (°F)	25.2
Flow rate (gal/min)	631,100
Heat transfer (Btu/hr)	$7{,}628 \times 10^6$ (note 3)
Wind velocity design (mph)	110
Seismic design criteria per International Building Code	

Notes:

1. This table supplements DCD Table 10.4.5-1.

2. Nominal CWS pump flow rate includes blowdown (14,440 gpm nominal) which is discharged from the pump discharge common header.

3. The inlet temperature (°F) is approximately 114.9°F, as the condenser outlet flow is expected to be cooled by the return flow stream from the TCS.

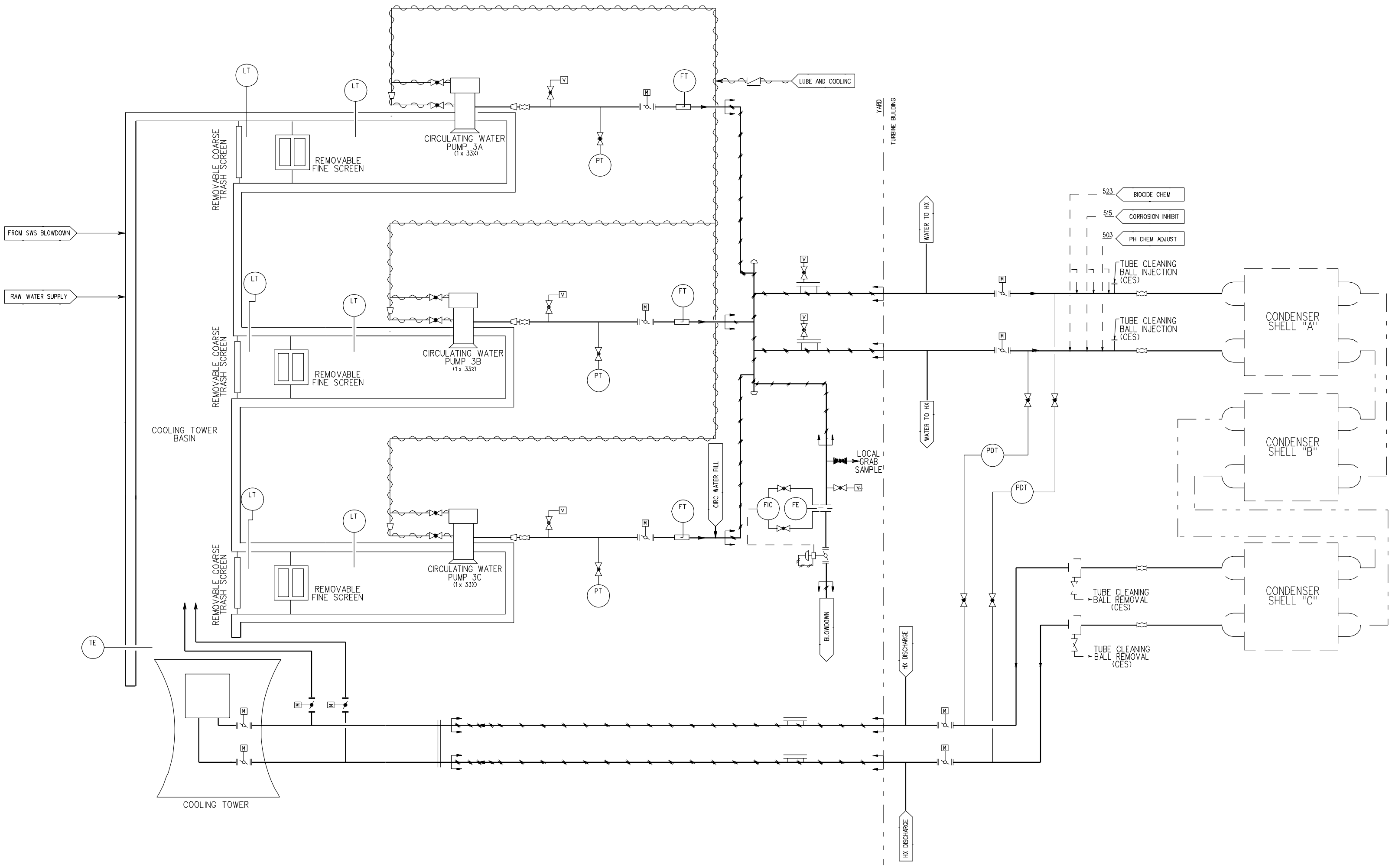

Figure 10.4-201
Circulating Water System Diagram

CHAPTER 11

RADIOACTIVE WASTE MANAGEMENT

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CHAPTER 11
RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

11.2.1.2.4 Controlled Release of Radioactivity

Add the following to the end of DCD Subsection 11.2.1.2.4:

VEGP SUP 11.2-1 The liquid radwaste system (WLS) exterior discharge piping from the Units 3 and 4 Radwaste Building is buried, stainless steel, enclosed within a guard pipe and monitored for leakage to comply with 10 CFR 20.1406. The WLS discharge lines connect to the Waste Water System (WWS) plant outfall pipe within the Exclusion Area Boundary for dilution below the release limits of 10 CFR Part 20, Appendix B, Table II, Column 2. Dilution at this point, downstream of the WWS blowdown sump, is primarily supplied by the circulating water blowdown flow. The blowdown sump and plant outfall are described in Subsection 9.2.9.2.2.

The WWS blowdown line to the plant outfall at the Savannah River is a high density polyethylene single-walled buried pipe. There are no valves, vacuum breakers, or pumps along the WWS blowdown line between the point where the WLS connects and the plant outfall. Monitoring for leakage downstream of the WLS radwaste discharge line connection is per NEI 08-08A (Reference 201) as described in Appendix 12AA. This monitoring will be implemented as part of the Units 3 and 4 groundwater monitoring program.

11.2.1.2.5.2 Use of Mobile and Temporary Equipment

Add the following information at the end of DCD Subsection 11.2.1.2.5.2:

STD COL 11.2-1 When mobile or temporary equipment is selected to process liquid effluents, the equipment design and testing meets the applicable requirements of Regulatory Guide 1.143. When confirmed through sampling that the radioactive waste contents do not exceed the A_2 quantities for radionuclides specified in Appendix A to 10 CFR Part 71, the liquid effluent may be processed with mobile or temporary equipment in the Radwaste Building. When the A_2 quantities are exceeded, liquid effluent is processed in the Seismic Category I auxiliary building.

Mobile and temporary equipment are designed in accordance with the applicable mobile and temporary radwaste treatment systems guidance provided in

Regulatory Guide 1.143, including the codes and standards listed in Table 1 of the Regulatory Guide.

Mobile and temporary equipment has the following features:

- Level indication and alarms (high-level) on tanks.
- Screwed connections are permitted only for instrument connections beyond the first isolation valve.
- Remote operated valves are used where operations personnel would be required to frequently manipulate a valve.
- Local control panels are located away from the equipment, in low dose areas.
- Instrumentation readings are accessible from the local control panels (i.e., temperature, flow, pressure, liquid level, etc.).
- Wetted parts are 300 series stainless steel, except flexible hose and gaskets.
- Flexible hose is used only for mobile equipment within the designated “black box” locations between mobile components and at the interface with the permanent plant piping.
- The contents of tanks are capable of being mixed, either through recirculation or with a mixer.
- Grab sample points are located in tanks and upstream and downstream of the process equipment.

Inspection and testing of mobile or temporary equipment is in accordance with the codes and standards listed in Table 1 of Regulatory Guide 1.143 with the following additions:

- After placement in the station, the mobile or temporary equipment is hydrostatically, or pneumatically, tested prior to tie-in to permanent plant piping.
- A functional test, using demineralized water, is performed. Remote operated valves are stroked (open-closed-open or closed-open-closed) under full flow conditions. The proper function of the instrumentation, including alarms, is verified. The operating procedures are verified correct during the functional test.
- Tank overflows are routed to floor drains.

- Floor drains are confirmed to be functional prior to placing mobile or temporary equipment into operation.

11.2.2.1.6 Prevention of Commingling of Chelating Agents With Radioactive Liquids

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VEGP ESP COL 2.4-1 Chelating agents, as defined in 10 CFR 61.2, are not routinely used in liquid radioactive waste processing at VEGP Units 1 and 2, and similarly, they will not be routinely used in liquid radioactive processing at VEGP Units 3 and 4. In the event chelating agents are required for a specific purpose (such as cleaning of steam generators or other plant systems), an evaluation will be conducted prior to use, and specific controls will be implemented to ensure that wastes are segregated and managed appropriately to prevent commingling with plant's normal liquid radwaste system.

11.2.3 RADIOACTIVE RELEASES

Add the following new paragraph at the end of DCD Subsection 11.2.3:

VEGP SUP 11.2-2 The only liquid effluent site interface parameter outside of the Westinghouse scope is the release point to the Savannah River.

11.2.3.3 Dilution Factor

Add the following information at the end of DCD Subsection 11.2.3.3.

VEGP COL 11.2-2 The site-specific dilution factor is addressed in Subsection 11.2.3.5.

11.2.3.5 Estimated Doses

Replace DCD Subsection 11.2.3.5 with the following.

VEGP COL 11.5-3 Subsection 11.2.3 of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

Add the following at the end of ESPA SSAR Subsection 11.2.3.2.

VEGP COL 11.2-2 ESPA SSAR Table 11.2-7 reports a total body population dose from liquid effluents within 50 miles of VEGP Units 3 and 4 of 0.037 person-rem/year or 0.019 person-rem/year per reactor. In addition, the corresponding thyroid dose has been calculated to be 0.0022 person-rem/year per reactor.

11.2.3.5.1 Liquid Radwaste Cost Benefit Analysis Methodology

STD COL 11.2-2 The application of the methodology of Regulatory Guide 1.110 was used to satisfy the cost benefit analysis requirements of 10 CFR Part 50. Appendix I, Section II.D. The parameters used in calculating the Total Annual Cost (TAC) are fixed and are given for each radwaste treatment system augment listed in Regulatory Guide 1.110, including the Annual Operating Cost (AOC) (Table A-2), Annual Maintenance Cost (AMC) (Table A-3), Direct Cost of Equipment and Materials (DCEM) (Table A-1), and Direct Labor Cost (DLC) (Table A-1). The following variable parameters were used:

- Capital Recovery Factor (CRF) -This factor is taken from Table A-6 of Regulatory Guide 1.110 and reflects the cost of money for capital expenditures. A cost-of-money value of 7% per year is assumed in this analysis, consistent with the "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (NUREG/BR-0058). A CRF of 0.0806 was obtained from Table A-6.

- Indirect Cost Factor (ICF) -This factor takes into account whether the radwaste system is unitized or shared (in the case of a multi-unit site) and is taken from Table A-5 of Regulatory Guide 1.110. It is assumed that the radwaste system for this analysis is a unitized system at a 2-unit site, which equals an ICF of 1.625.

- Labor Cost Correction Factor (LCCF) -This factor takes into account the differences in relative labor costs between geographical regions and is taken from Table A-4 of Regulatory Guide 1.110. A LCCF of 1.0 (the lowest value) is assumed in this analysis.

Appendix I to 10 CFR Part 50 prescribes a $1,000 per person-rem criterion for determining the cost benefit of actions to reduce radiation exposure.

The analysis used a conservative assumption that the respective radwaste treatment system augment is a "perfect" system that reduces the effluent and dose by 100 percent. The liquid radwaste treatment system augments annual costs were determined and the lowest annual cost considered a threshold value. The lowest-cost option for liquid radwaste treatment system augments is a 20 gpm Cartridge Filter at $11,140 per year, which yields a threshold value of 11.14 person-rem total body or thyroid dose from liquid effluents.

For AP1000 sites with population dose estimates less than 11.14 person-rem total body or thyroid dose from liquid effluents. no further cost-benefit analysis is needed to demonstrate compliance with 10 CFR 50, Appendix I Section II.D.

11.2.3.5.2 Liquid Radwaste Cost Benefit Analysis

VEGP COL 11.2-2 As discussed in Section 11.2.3.5.1, the lowest cost liquid radwaste system augment is $11,140. Assuming 100% efficiency of this augment. the minimum possible cost per person-rem is determined by dividing the cost of the augment by the population dose. This is $586,316 per person-rem total body ($11,140/0.019 person-rem) and $5,063,636 per person-rem thyroid ($11,140/0.0022 person-rem). These costs per person-rem reduction exceed the $1,000 per person-rem criterion prescribed in Appendix I to 10 CFR Part 50 and are therefore not beneficial.

11.2.3.6 Quality Assurance

STD SUP 11.2-1 Add the following to the end of DCD Subsection 11.2.3.6:

Since the impact of radwaste systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. Thus, a supplemental quality assurance program applicable to design, construction, installation and testing provisions of the liquid radwaste system is established by procedures that complies with the guidance presented in Regulatory Guide 1.143.

11.2.5 COMBINED LICENSE INFORMATION

11.2.5.1 Liquid Radwaste Processing by Mobile Equipment

STD COL 11.2-1 This COL Item is addressed in Subsection 11.2.1.2.5.2.

11.2.5.2 Cost Benefit Analysis of Population Doses

STD COL 11.2-2 This COL item is addressed in Subsection 11.2.3.5.1.

VEGP COL 11.2-2 This COL Item is addressed in Subsections 11.2.3.3, 11.2.3.5 and 11.2.3.5.2.

11.2.6 REFERENCES

201. NEI 08-08A, Generic FSAR Template Guidance for Life Cycle Minimization of Contamination, Revision 0, October 2009 (ML093220445).

11.3 GASEOUS WASTE MANAGEMENT SYSTEM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

11.3.3 RADIOACTIVE RELEASES

Add the following new paragraph at the end of DCD Subsection 11.3.3:

STD SUP 11.3-2 There are no gaseous effluent site interface parameters outside of the Westinghouse scope.

11.3.3.4 Estimated Doses

Replace DCD Subsection 11.3.3.4 with the following.

VEGP COL 11.5-3 Subsection 11.3.3 of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

Add the following at the end of ESPA SSAR Subsection 11.3.3.2.

VEGP COL 11.3-1 ESPA SSAR Table 11.3-8 reports a total body population dose from gaseous effluents within 50 miles of VEGP Units 3 and 4 of 1.8 person-rem/year or 0.9 person-rem/per reactor. In addition, the corresponding thyroid dose has been calculated to be 3.0 person-rem/year per reactor.

11.3.3.4.1 Gaseous Radwaste Cost-Benefit Analysis Methodology

STD COL 11.3-1 The guidance for performing cost-benefit analysis for the gaseous radwaste system is similar to that used and described for the liquid radwaste system in Section 11.2. The gaseous radwaste treatment system augments annual costs were determined and the lowest annual cost considered a threshold value. The lowest-cost option for gaseous radwaste treatment system augments is the Steam Generator Flash Tank Vent to Main Condenser at $6,320 per year, which yields a threshold value of 6.32 person-rem total body or thyroid from gaseous effluents.

For AP1000 sites with population dose estimates less than 6.32 person-rem total body or thyroid dose from gaseous effluents, no further cost-benefit analysis is needed to demonstrate compliance with 10 CFR 50, Appendix I, Section II.D.

11.3.3.4.2 Gaseous Radwaste Cost-Benefit Analysis

VEGP COL 11.3-1 As discussed in Section 11.3.3.4.1, the lowest cost gaseous radwaste system augment is $6,320. Assuming 100 percent efficiency of this augment, the minimum possible cost per person-rem is determined by dividing the cost of the augment by the population dose. This is $7,022 per person-rem total body ($6,320/0.9 person-rem) and $2,107 per person-rem thyroid ($6,320/3.0 person-rem thyroid). These costs per person-rem reduction exceed the $1,000 per person-rem criterion prescribed in Appendix I to 10 CFR Part 50 and are therefore not cost beneficial.

11.3.3.6 Quality Assurance

Add the following to the end of DCD Subsection 11.3.3.6:

STD SUP 11.3-1 Since the impact of radwaste systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. Thus, a supplemental quality assurance program applicable to design, construction, installation, and testing provisions of the gaseous radwaste system is established by procedures that complies with the guidance presented in Regulatory Guide 1.143.

11.3.5 COMBINED LICENSE INFORMATION

11.3.5.1 Cost Benefit Analysis of Population Doses

STD COL 11.3-1 This COL Item is addressed in Subsection 11.3.3.4.1.

VEGP COL 11.3-1 This COL Item is addressed in Subsections 11.3.3.4 and 11.3.3.4.2.

11.3.6 REFERENCES

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201. Deleted.

11.4 SOLID WASTE MANAGEMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following after DCD Subsection 11.4.2.4.2:

11.4.2.4.3 Alternatives for B and C Wastes

VEGP COL 11.4-1 It is expected that Class B and C wastes will constitute approximately 5 percent by volume of the low level radioactive waste (LLRW) that will be generated by the plant with the balance being Class A waste. The volume of wet Class B and C waste is approximately 100 percent of the total Class B and C waste. As of July 1, 2008, the LLRW disposal facility in Barnwell, South Carolina is no longer accepting Class B and C waste from sources in states that are outside of the Atlantic Compact. However, the disposal facility in Clive, Utah is still accepting Class A waste from out of state. Should there be no disposal facilities that will accept the Class B and C wastes after the plant begins operation, there are several options available for storage of such waste:

- As provided in referenced DCD Subsection 11.4.2., the Auxiliary Building is designed to have more than a year of spent resin storage capacity at the expected rate and the spent resin tanks may be mixed to limit the radioactivity concentrations thereby limiting the volume of Class B and C wet waste requiring storage.
- Vendor services are available to process Class A, B, and C waste and transfer for storage of that material until a disposal site is available. Currently, Waste Control Specialists (WCS) of Texas is available to store Class A, B, and C material pending the availability of a licensed disposal site.
- If additional storage capacity were eventually needed, the plant could construct or expand storage facilities onsite or gain access to a storage facility at another licensed nuclear plant.

11.4.5 QUALITY ASSURANCE

Add the following to the end of DCD Subsection 11.4.5:

STD SUP 11.4-1 Since the impact of radwaste systems on safety is limited, the extent of control required by Appendix B to 10 CFR Part 50 is similarly limited. Thus, a supplemental quality assurance program applicable to design, construction, installation and testing provisions of the solid radwaste system is established by procedures that complies with the guidance presented in Regulatory Guide 1.143.

11.4.6 COMBINED LICENSE INFORMATION FOR SOLID WASTE MANAGEMENT SYSTEM PROCESS CONTROL PROGRAM

Add the following information to the end of DCD Subsection 11.4.6.

This COL Item is addressed below.

STD COL 11.4-1 A Process Control Program (PCP) is developed and implemented in accordance with the recommendations and guidance of NEI 07-10A (Reference 201). 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. Its purpose is to provide 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 (LLW) disposal site that is licensed in accordance with 10 CFR Part 61.

Waste processing (solidification or dewatering) equipment and services may be provided by the plant or by third-party vendors. Each process used meets the applicable requirements of the PCP.

No additional onsite radwaste storage is required beyond that described in the DCD.

Table 13.4-201 provides milestones for PCP implementation.

11.4.6.1 Procedures

STD SUP 11.4-1 Operating procedures specify the processes to be followed to ship waste that complies with the waste acceptance criteria (WAC) of the disposal site, 10 CFR 61.55 and 61.56, and the requirements of third party waste processors.

Each waste stream process is controlled by procedures that specify the process for packaging, shipment, material properties, destination (for disposal or further processing), testing to verify compliance, the process to address non-conforming materials, and required documentation.

Where materials are to be disposed of as non-radioactive waste (as described in DCD Subsection 11.4.2.3.3), final measurements of each package are performed to verify there has not been an accumulation of licensed material resulting from a buildup of multiple, non-detectable quantities. These measurements are obtained using sensitive scintillation detectors, or instruments of equal sensitivity, in a low-background area.

Procedures document maintenance activities, spill abatement, upset condition recovery, and training.

Procedures document the periodic review and revision, as necessary, of the PCP based on changes to the disposal site, WAC regulations, and third party PCPs.

11.4.6.2 Third Party Vendors

Third party equipment suppliers and/or waste processors are required to supply approved PCPs. Third party vendor PCPs describe compliance with Regulatory Guide 1.143, Generic Letter 80-09, and Generic Letter 81-39. Third party vendor PCPs are referenced appropriately in the plant PCP before commencement of waste processing.

11.4.6.3 Long Term On-Site Storage Facility

VEGP SUP 11.4-1 Storage space for six-month's volume of packaged waste is provided in the radwaste building. Radioactive waste generated by VEGP Units 3 and 4 will normally be shipped to a licensed disposal or off-site storage facility. However, should disposal facilities or off-site storage facilities not be available, storage capacity will be expanded as described below to provide additional on-site storage for VEGP Units 3 and 4.

Additional on-site low-level radioactive waste (LLRW) storage capabilities are available if Class B and C waste cannot be disposed at a licensed disposal facility. An outside storage pad will be utilized to provide this capability. The VEGP Units 3 and 4 LLRW storage facility would be located outside the Protected Area (PA) in the Owner Controlled Area (OCA). The storage facility would be enclosed by an eight-foot high fence with locked gates and would be provided with area lighting. The storage of LLRW would be in high integrity containers (HICs) or other suitable containers that will not decay over time, which would be stored within shielded containers. The design of the storage facility will comply with the guidance of documents as identified in this section which is consistent with NUREG-0800, Appendix 11.4A. The design storage capacity is based on the expected generation in Table 11.4-1, industry experience that indicates approximately 100% of the Class B and C waste is expected to be in the form of wet waste, and volume minimization/reduction programs. The site waste management plan will include radioactive wet waste reduction initiatives for waste Class B and C.

The storage facility will be sited such that it could be sized to accommodate storage of Class B and C waste over the operating life of the plant and designed to accommodate future expansion as needed. Capacity would be added in phases based on the expected availability of off-site treatment and storage, and disposal facilities.

11.4.6.3.1 Outside Storage Pad Design Considerations

The following design considerations would be applied to the on-site LLRW storage facility: (References 202, 203, and 204):

- The location of the storage pad would meet the dose rate criteria of 40 CFR 190 and 10 CFR 20.1302 for both the site boundary and unrestricted area. The onsite storage will be located such that any additional dose contributes less than 1 mrem per year to the 40 CFR Part 190 limits. Onsite dose limits will be controlled per 10 CFR 20, including the ALARA principle of 10 CFR 20.1101.

- The outside storage pad would be an engineered feature designed to minimize settling and would be constructed of reinforced concrete or engineered gravel.

- The storage pad location would avoid natural or engineered surface drainage and be located at an elevation with regard to the site's design bases flood level.

- The storage pad would have a fence or other suitable security measures consistent with its location on the site.

- The waste containers (typically high integrity containers) would be stored inside of a shielded container, typically consisting of reinforced concrete containers that provide radiation shielding and weather protection.

- The configuration of the storage shields would be arranged to be accessible from the perimeter road or from a center aisle using a mobile crane (if used).

- Personnel passages would be provided between rows of storage shields for access to the container for inspection.

- Adequate electrical power and lighting would be provided at the storage facility to allow power for tools, analytical equipment, sample pumps, radiation instruments, boroscope lights, etc.

- Fire protection, fire hydrants or fire extinguishers, for vehicle fires should be provided.

11.4.6.3.2 Outside Storage Pad Operating Considerations

The following operating considerations for on-site storage pad operations are based on NRC and Industry guidance (References 202, 203, and 204) and would be included in operating procedures:

- Identification of the arrangement of storage shields, waste handling, storage methods, safety analysis limitations, accident conditions, and off site dose calculations.

- The use of hold-down devices to secure the waste container during severe environmental events, such as strong wind would be provided for, unless the waste container and storage shields can be demonstrated to remain in place without restraints during such events.

- The waste container selected for use is compatible with the waste form stored to ensure waste container integrity.

- Shielding requirements would be determined before the waste container is loaded into a storage shield to eliminate the radiation exposure associated with adding additional shielding.

- If additional shield walls around the perimeter of the storage pad are required, the shield walls would be easily installed and capable of being moved.

- Periodic inspection and testing requirements for outside storage pad operation would include the following:

    - Dose rate and contamination surveys in accordance with health physics procedures.

    - Sampling of storage shields for water and storage shields containing dewatered resin for explosive gas build-up.

    - Visual inspection of selected waste containers in storage to detect unexpected changes / container integrity. (Remote inspection methods and the use of high integrity containers will allow reduced scope for ALARA practices.)

    - Defoliation and general condition of the onsite storage pad.

- Total radioactive material inventory limits would be established to demonstrate compliance with the design limits for the storage area, dose limits for members of the public and safety features or measures provided by the storage module.

- The contents of records for inventory controls, monitoring and inspection and other relevant data are maintained and retrievable.

- Operational safety features for handling waste containers and storage shields would include the training required for personnel operating cranes, forklifts, tie downs and heavy equipment during any waste container/ storage shield transfer activity.

- Criteria for the end of storage period that would include waste container inspection and additional reprocessing required prior to shipment offsite.

11.4.7 REFERENCES

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201. NEI 07-10A, "Generic FSAR Template Guidance for Process Control Program (PCP)," Revision 0, March 2009 (ML091460627).

202. Technical Report 1018644 "Guidelines for Operating an Interim On Site Low Level Radioactive Waste Storage Facility," Revision 1, EPRI, Palo Alto, CA, February 2009.

203. Regulatory Issue Summary 2008-32 "Interim Low Level Radioactive Waste Storage at Reactor Sites," December 2008.

204. Generic Letter (GL) 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites," November 1981.

11.5 RADIATION MONITORING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

11.5.1.2 Power Generation Design Basis

Revise the fourth bullet in DCD Subsection 11.5.1.2 as follows:

STD COL 11.5-2

- Data collection and data storage to support compliance reporting for the applicable NRC requirements and guidelines, such as General Design Criterion 64 and Regulatory Guide 1.21 and Regulatory Guide 4.15, Revision 1.

11.5.2.4 Inservice Inspection, Calibration, and Maintenance

Add the following information at the end of DCD Subsection 11.5.2.4:

STD COL 11.5-2 Daily checks of effluent monitoring system operability are made by observing channel behavior. Detector response is routinely observed with a remotely-positioned check source in accordance with plant procedures. Instrument background count rate is also observed to determine proper functioning of the monitors. Any detector whose response cannot be verified by observation during normal operation or by using the remotely-positioned check source can have its response checked with a portable check source. A record is maintained showing the background radiation level and the detector response.

Calibration of the continuous radiation monitors is done with commercial radionuclide standards that have been standardized using a measurement system traceable to the National Institute of Standards and Technology.

11.5.3 EFFLUENT MONITORING AND SAMPLING

Add the following information at the end of the DCD Subsection 11.5.3.

VEGP COL 11.5-2 SNC is extending the existing SNC program for quality assurance of radiological effluent and environmental monitoring that is based on Regulatory Guide 4.15, Revision 1, to apply to Vogtle Units 3 and 4. Regulatory Guide 4.15, Revision 1, is a proven methodology for quality assurance of radiological effluent and environmental monitoring programs that is acceptable to the NRC staff as a method for demonstrating compliance with applicable requirements of 10 CFR Parts 20, 50, 52, 61, and 72. Use of Revision 2 of Regulatory Guide 4.15 would necessitate conducting two separate programs involving the use of common staff facilities, and equipment, which will create an undue burden and may lead to an increased possibility for human error. Therefore, SNC commits to use Regulatory Guide 4.15, Revision 1, methodology for Vogtle Units 3 and 4 for optimal consistency, efficiency, and practicality.

11.5.4 PROCESS AND AIRBORNE MONITORING AND SAMPLING

STD COL 11.5-2 Add the following information at the end of the first paragraph in DCD Subsection 11.5.4.

The sampling program for liquid and gaseous effluents will conform to Regulatory Guide 4.15, Revision 1 (See Appendix 1AA).

Add the following information at the end of DCD Subsection 11.5.4.

11.5.4.1 Effluent Sampling

STD COL 11.5-2 Effluent sampling of potential radioactive liquid and gaseous effluent paths is conducted on a periodic basis to verify effluent processing meets the discharge limits to offsite areas. The effluent sampling program provides the information for the effluent measuring and reporting required by 10 CFR 50.36a and 10 CFR Part 20 and implemented through the Offsite Dose Calculation Manual (ODCM) and plant procedures. The frequency of the periodic sampling and analyses described herein are nominal and may be increased as permitted by procedure. Tables 11.5-201 and 11.5-202 summarize the sample and analysis schedules and sensitivities, respectively. The information contained in Tables 11.5-201 and 11.5-202 are derived from Regulatory Guide 1.21.

Laboratory isotopic analyses are performed on continuous and batch effluent releases in accordance with the ODCM. Results of these analyses are compiled and appropriate portions are utilized to produce the Radioactive Effluent Release Report.

11.5.4.2 Representative Sampling

Representative samples are obtained from well-mixed streams or volumes of effluent liquid through the use of proper sampling equipment, proper location of sampling points, and the development and use of sampling procedures. The recommendations of ANSI N 42.18 (Reference 203) are considered for the selection of instrumentation specific to the continuous monitoring of radioactivity in liquid effluents.

Sampling of effluent liquids is consistent with guidance in Regulatory Guide 1.21. When practical, effluent releases are batch-controlled, and prior to sampling, large volumes of liquid waste are mixed, in as short a time span as practicable, so that solid particulates are uniformly distributed in the liquid volume. Sampling and analysis is performed, and release conditions set, before release. Sample points are located to minimize flow disturbance due to fittings and other characteristics of equipment and components. Sample lines are flushed consistent with plant procedures to remove sediment deposits.

Representative sampling of process effluents is attained through sample and monitor locations and methods and criteria detailed in plant procedures.

Composite sampling is employed to analyze for hard to measure radionuclides and to monitor effluent streams that normally are not expected to contain significant amounts of radioactive contamination. Composite liquid samples are collected in proportion to the volume of each batch of effluent release. The composite is thoroughly mixed prior to analysis. Collection periods for composites are as short as practicable and periodic checks are performed to identify changes in composite samples. When grab samples are collected instead of composite samples, the time of the sample, location, and frequency are considered to provide a representative sample of the radioactive materials.

The pressure head of the fluid, if available, is used for taking samples. If sufficient pressure head is not available to take samples, then sample pumps are used to draw the sample from the process fluid to the detector panels and back to the process.

Testing and obtaining representative samples using the radiation monitors described in DCD Subsection 11.5 will be performed in accordance with ANSI N13.1 (Reference 201).

For obtaining representative samples in unfiltered ducts, isokinetic probes are tested and used in accordance with ANSI N13.1 (Reference 201).

Analytical Procedures

Typically, samples of process and effluent gases and liquids are analyzed in the station laboratory or by an outside laboratory via the following techniques:

- Gross alpha/beta counting
- Gamma spectrometry
- Liquid scintillation counting

"Available" instrumentation and counting techniques change as other instruments and techniques become available. For this reason, the frequency of sampling and the analysis of samples are generalized in this subsection.

Gross alpha/beta analysis may be performed directly on unprocessed samples (e.g., air filters) or on processed samples (e.g., evaporated liquid samples). Sample volume, counting geometry, and counting time are chosen to match measurement capability with sample activity. Correction factors for sample-detector geometry, self-absorption and counter resolving time are applied to provide the required accuracy.

Liquid effluent samples are prepared for alpha/beta counting by evaporation onto steel planchets. Gamma analysis may be done on any type of sample (gas, solid or liquid) in a gamma spectrometer.

Tritiated water vapor samples are collected by condensation or adsorption, and the resultant liquid is analyzed by liquid scintillation counting techniques.

Radiochemical separations are used for the routine analysis of Sr-89 and Sr-90.

Liquid samples are collected in polyethylene bottles to minimize absorption of nuclides onto container walls.

11.5.6.5 Quality Assurance

Add the following information at the end of DCD Subsection 11.5.6.5.

STD COL 11.5-2 The sampling program and the associated monitors conform to Regulatory Guide 4.15, Revision 1 (See Appendix 1AA).

11.5.8 COMBINED LICENSE INFORMATION

STD COL 11.5-1 An Offsite Dose Calculation Manual (ODCM) is developed and implemented in accordance with the recommendations and guidance of NEI 07-09A (Reference 202). The ODCM contains the methodology and parameters used for calculating doses resulting from liquid and gaseous effluents. The ODCM addresses operational setpoints, including planned discharge rates, for radiation monitors and monitoring programs (process and effluent monitoring and environmental monitoring) for the control and assessment of the release of radioactive material to the environment. The ODCM provides 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. The ODCM will be finalized prior to fuel load with site-specific information.

Table 13.4-201 provides milestones for ODCM implementation.

STD COL 11.5-2 This COL Item is addressed in Subsections 11.5.1.2, 11.5.2.4, 11.5.4, 11.5.4.1, 11.5.4.2, and 11.5.6.5.

VEGP COL 11.5-2 This COL Item is addressed in Subsection 11.5.3.

VEGP COL 11.5-3 This COL Item is addressed in Subsections 11.2.3.5 and 11.3.3.4 for liquid and gaseous effluents, respectively.

Add the following subsection after DCD Subsection 11.5.8.

11.5.9 REFERENCES

201. ANSI N13.1-1969, “Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities.”

202. NEI 07-09A, “Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description,” Revision 0, March 2009 (ML091050234).

203. ANSI N42.18-2004, “Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents.”

STD COL 11.5-2

Table 11.5-201 (Sheet 1 of 2)
Minimum Sampling Frequency

Stream	Sampled Medium	Frequency
Gaseous	Continuous Release	A sample is taken within one month of initial criticality, and at least weekly thereafter to determine the identity and quantity for principal nuclides being released. A similar analysis of samples is performed following each refueling, process change, or other occurrence that could alter the mixture of radionuclides. When continuous monitoring shows an unexplained variance from an established norm. Monthly for tritium.
	Batch Release	Prior to release to determine the identity and quantity of the principal radionuclides (including tritium).
	Filters (particulates)	Weekly. Quarterly for Sr-89 and Sr-90. Monthly for gross alpha.

STD COL 11.5-2

Table 11.5-201 (Sheet 2 of 2)
Minimum Sampling Frequency

Stream	Sampled Medium	Frequency
Liquid	Continuous Releases	Weekly for principal gamma-emitting radionuclides. Monthly, a composite sample for tritium and gross alpha. Monthly, a representative sample for dissolved and entrained fission and activation gases. Quarterly, a composite sample for Sr-89, Sr-90, and Fe-55.
	Batch Releases	Prior to release for principal gamma-emitting radionuclides. Monthly, a composite sample for tritium and gross alpha. Monthly, a representative sample from at least one representative batch for dissolved and entrained fission and activation gases. Quarterly, a composite sample for Sr-89, Sr-90 and Fe-55.

STD COL 11.5-2

Table 11.5-202
Minimum Sensitivities

Stream	Nuclide	Sensitivity
Gaseous	Fission & Activation Gases	1.0E-04 µCi/cc
	Tritium	1.0E-06 µCi/cc
	Iodines & Particulates	Sufficient to permit measurement of a small fraction of the activity that would result in annual exposures of 15 mrem to thyroid for iodines, and 15 mrem to any organ for particulates, to an individual in an unrestricted area.
	Gross Radioactivity	Sufficient to permit measurement of a small fraction of the activity that would result in annual air dose of 1) 10 mrad due to gamma, and 2) 20 mrad of beta at any location near ground level at or beyond the site boundary.
Liquid	Gross Radioactivity	1.0E-07 µCi/ml
	Gamma-emitters	5.0E-07 µCi/ml
	Dissolved & Entrained Gases	1.0E-05 µCi/ml
	Gross Alpha	1.0E-07 µCi/ml
	Tritium	1.0E-05 µCi/ml
	Sr-89 & Sr-90	5.0E-08 µCi/ml
	Fe-55	1.0E-06 µCi/ml

CHAPTER 12

RADIATION PROTECTION

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CHAPTER 12
RADIATION PROTECTION

12.1 ASSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY ACHIEVABLE (ALARA)

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD COL 12.1-1 This section incorporates by reference NEI 07-08A, Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA), Revision 0. 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.

Revise the last sentence of NEI 07-08A Subsection 12.1.2 to read:

STD COL 12.1-1 ALARA procedures are established, implemented, maintained and reviewed consistent with 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description, which is discussed in Section 17.5.

Add the following information at the end of DCD Subsection 12.1.2.4:

12.1.2.4.3 Equipment Layout

STD SUP 12.1-1 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.

12.1.3 COMBINED LICENSE INFORMATION

STD COL 12.1-1 This COL item is addressed in NEI 07-08A and Appendix 12AA.

12.2 RADIATION SOURCES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

12.2.1.1.10 Miscellaneous Sources

Add the following information at the end of DCD Subsection 12.2.1.1.10:

STD COL 12.2-1 Licensed sources containing byproduct, source, and special nuclear material that warrant shielding design consideration meet the applicable requirements of 10 CFR Parts 20, 30, 31, 32, 33, 34, 40, 50, and 70.

There are byproduct and source materials with known isotopes and activity manufactured for the purpose of measuring, checking, calibrating, or controlling processes quantitatively or qualitatively.

These sources include but are not limited to:

- Sources in field monitoring equipment.
- Sources in radiation monitors to maintain a threshold sensitivity.
- Sources used for radiographic operations.
- Depleted uranium slabs used to determine beta response and correction factors for portable monitoring instrumentation.
- Sources used to calibrate and response check field monitoring equipment (portable and fixed).
- Liquid standards and liquids or gases used to calibrate and verify calibration of laboratory counting and analyzing equipment.
- Radioactive waste generated by the use of radioactive sources.

Specific details of these sources are maintained in a database on-site following procurement. This database, at a minimum, contains the following information:

- Isotopic composition
- Location in the plant
- Source strength

- Source geometry

Written procedures are established and implemented that address procurement, receipt, inventory, labeling, leak testing, surveillance, control, transfer, disposal, storage, issuance and use of these radioactive sources. These procedures are developed in accordance with the radiation protection program to comply with 10 CFR Parts 19 and 20. A supplementary warning symbol is used in the presence of large sources of ionizing radiation consistent with the guidance in Regulatory Issue Summary (RIS) 2007-03.

Sources maintained on-site for instrument calibration purposes are shielded while in storage to keep personnel exposure ALARA. Sources used to service or calibrate plant instrumentation are also routinely brought on-site by contractors. Radiography is performed by the licensed utility group or licensed contractors. These sources are maintained and used in accordance with the provisions of the utility group's or contractor's license. Additional requirements and restrictions may apply depending on the type of source, use, and intended location of use. If the utility group or contractor source must be stored on-site, designated plant personnel must approve the storage location, and identify appropriate measures for maintaining security and personnel protection.

During the period prior to the implementation of the Emergency Plan (in preparation for the initial fuel loading following the 52.103(g) finding), no specific materials related emergency plan will be necessary because:

a) No byproduct material will be received, possessed, or used in a physical form that is "in unsealed form, on foils or plated sources, or sealed in glass," that exceeds the quantities in Schedule C in 10 CFR 30.72, and

b) No 10 CFR Part 40 specifically licensed source material, including natural uranium, depleted uranium and uranium hexafluoride will be received, possessed, or used during this period.

The following radioactive sources will be used for the Radiation Monitoring System and laboratory/portable monitoring instrumentation:[1]

Radioactive Licensee Material (Element and Mass Number)[1]	Chemical and/or Physical Form[1]	Maximum Quantity That Licensee May Possess at Any One Time[1]
• Any byproduct material with atomic numbers 1 through 93 inclusive	Sealed Sources[2]	No single source to exceed 100 millicuries 5 Curies total
• Americium-241	Sealed Sources[2]	No single source to exceed 300 millicuries 500 millicuries total

Notes:
1. This information remains in effect between the issuance of the COL and the Commission's 52.103(g) finding for each unit, and will be designated historical information after that time.
2. Includes calibration and reference sources.

12.2.3 COMBINED LICENSE INFORMATION

STD COL 12.2-1 This COL item is addressed in Subsection 12.2.1.1.10.

12.3 RADIATION PROTECTION DESIGN FEATURES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

12.3.1 FACILITY DESIGN FEATURES

12.3.1.2 Radiation Zoning and Access Control

VEGP DEP 18.8-1 Add the following information at the end of the second paragraph in DCD Subsection 12.3.1.2.

Figure 12.3-201, Figure 12.3-202, and Figure 12.3-203 replace DCD Figure 12.3-1 (sheet 11), DCD Figure 12.3-2 (sheet 11), and DCD Figure 12.3-3 (sheet 11), respectively, to reflect the relocation of the Operations Support Center.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

Add the following text to the end of DCD Subsection 12.3.4.

STD COL 12.3-2 Procedures detail 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 Regulatory Guides 1.21-Appendix A, 8.2, 8.8, and 8.10. Additional discussion of radiological surveillance practices is included in the radiation protection program description provided in Appendix 12AA.

Surveillance requirements are determined by the functional manager in charge of radiation protection based on actual or potential radiological conditions encountered by personnel and the need to identify and control radiation, contamination, and airborne radioactivity. These requirements are consistent with the operational philosophy in Regulatory Guide 8.10. Frequency of scheduled surveillance may be altered by permission of the functional manager in charge of radiation protection or their designee. Radiation Protection periodically provides cognizant personnel with survey data that identifies radiation exposure gradients in area resulting from identified components. This data includes recent reports, with survey data, location and component information.

The following are typical criteria for frequencies and types of surveys:

Job Coverage Surveys

- Radiation, contamination, and/or airborne surveys are performed and documented to support job coverage.
- Radiation surveys are sufficient in detail for Radiation Protection to assess the radiological hazards associated with the work area and the intended/ specified work scope.
- Surveys are performed commensurate with radiological hazard, nature and location of work being conducted.
- Job coverage activities may require surveys to be conducted on a daily basis where conditions are likely to change.

Radiation Surveys

- Radiation surveys are performed at least monthly in any radiological controlled area (RCA) where personnel may frequently work or enter. Survey frequencies may be modified by the functional manager in charge of radiation protection as previously noted.
- Radiation surveys are performed prior to or during entry into known or suspected high radiation areas for which up to date survey data does not exist.
- Radiation surveys are performed prior to work involving highly contaminated or activated materials or equipment.
- Radiation surveys are performed at least semiannually in areas outside the RCA. Areas to be considered include shops, offices, and storage areas.
- Radiation surveys are performed to support movement of highly radioactive material.
- Neutron radiation surveys are performed when personnel may be exposed to neutron emitting sources.

Contamination Surveys

- Contamination surveys are performed at least monthly in any RCA where personnel may frequently work or enter. Survey frequencies may be modified by the functional manager in charge of radiation protection as previously noted.

- Contamination surveys are performed during initial entry into known or suspected contamination area(s) for which up to date survey data does not exist.

- Contamination surveys are performed at least daily at access points, change areas, and high traffic walkways in RCAs that contain contaminated areas. Area access points to a High Radiation Area or Very High Radiation Area are surveyed prior to or upon access by plant personnel or if access has occurred.

- Contamination surveys are performed at least semiannually in areas outside the RCA. Areas to be considered include shops, offices, and storage areas.

- A routine surveillance is conducted in areas designated by the functional manager in charge of radiation protection or their designee likely to indicate alpha radioactivity. If alpha contamination is identified, frequency and scope of the routine surveillance is increased.

<u>Airborne Radioactivity Surveys</u>

- Airborne radioactivity surveys are performed during any work or operation in the RCA known or suspected to cause airborne radioactivity (e.g., grinding, welding, burning, cutting, hydrolazing, vacuuming, sweeping, use of compressed air, using volatiles on contaminated material, waste processing, or insulation).

- Airborne radioactivity surveys are performed during a breach of a radioactive system, which contains or is suspected of containing significant levels of contamination.

- Airborne radioactivity surveys are performed during initial entry (and periodically thereafter) into any known or suspected airborne radioactivity area.

- Airborne radioactivity surveys are performed immediately following the discovery of a significant radioactive spill or spread of radioactive contamination, as determined by the functional manager in charge of radiation protection.

- Airborne radioactivity surveys are performed daily in occupied radiological controlled areas where the potential for airborne radioactivity exists, including containment.

- Airborne radioactivity surveys are performed any time respiratory protection devices, alternative tracking methods such as derived air concentration-hour (DAC-hr), and/or engineering controls are used to control internal exposure.

- Airborne radioactivity surveys are performed using continuous air monitors (CAMs) for situations in which airborne radioactivity levels can fluctuate and early detection of airborne radioactivity could prevent or minimize inhalations of radioactivity by workers. Determination of air flow patterns are considered for locating air samplers.

- Airborne radioactivity surveys are performed prior to use and monthly during use on plant service air systems used to supply air for respiratory protection to verify the air is free of radioactivity.

- Tritium sampling is performed near the spent fuel pit when irradiated fuel is in the pit and other areas of the plant where primary system leaks occur and tritium is suspected.

Appropriate counting equipment is used based on the sample type and the suspected identity of the radionuclides for which the sample is being done. Survey results are documented, retrievable, and processed per site document control and records requirements consistent with Regulatory Guide 8.2. Completion of survey documentation includes the update of room/area posting maps and revising area or room postings and barricades as needed.

Air samples indicating activity levels greater than a procedure specified percentage of DAC are forwarded to the radiochemistry laboratory for isotopic analysis. Samples which cannot be analyzed on-site are forwarded to an offsite laboratory or a contractor for analysis; or, the DAC percentage may be hand calculated using appropriate values from 10 CFR Part 20, Appendix B.

The responsible radiation protection personnel review survey documentation to evaluate if surveys are appropriate and obtained when required, records are complete and accurate, and adverse trends are identified and addressed.

An in-plant radiation monitoring program maintains the capability to accurately determine the airborne iodine concentration in areas within the facility where personnel may be present under accident conditions. This program includes the training of personnel, procedures for monitoring, and provisions for maintenance of sampling and analysis equipment consistent with Regulatory Guides 1.21 (Appendix A) and 8.8. Training and personnel qualifications are discussed in Appendix 12AA.

A portable monitor system meeting the requirements of NUREG-0737, Item III.D.3.3, is available. The system uses a silver zeolite or charcoal iodine sample cartridge and a single-channel analyzer. The use of this portable monitor is incorporated in the emergency plan implementing procedures. The portable monitor is part of the in-plant radiation monitoring program. It is used to determine the airborne iodine concentration in areas where plant personnel may be present during an accident. Accident monitoring instrumentation complies with applicable parts of 10 CFR Part 50, Appendix A.

Sampling cartridges can be removed to a low background area for further analysis. These cartridge samples can be purged of any entrapped noble gases, when necessary, prior to being analyzed.

12.3.5.1 Administrative Controls for Radiological Protection

STD COL 12.3-1 This COL Item is addressed in Subsection 12.5.4 and Appendix 12AA.

12.3.5.2 Criteria and Methods for Radiological Protection

STD COL 12.3-2 This COL Item is addressed in Subsection 12.3.4.

12.3.5.3 Groundwater Monitoring Program

STD COL 12.3-3 This COL Item is addressed in Appendix 12AA.

12.3.5.4 Record of Operational Events of Interest for Decommissioning

STD COL 12.3-4 This COL Item is addressed in Appendix 12AA.

Security-Related Information — Withheld Under 10 CFR 2.390(d)
(See Part 9 of this COL Application)

(Note: This figure replaces DCD Figure 12.3-1 Sheet 11 of 16.)

VEGP DEP 18.8-1

Figure 12.3-201
Radiation Zones, Normal Operations /Shutdown
Annex Building, Elevation 100’-0” & 107’-2”

Security-Related Information — Withheld Under 10 CFR 2.390(d)
(See Part 9 of this COL Application)

(Note: This figure replaces DCD Figure 12.3-2 Sheet 11 of 15.)

VEGP DEP 18.8-1

Figure 12.3-202
Radiation Zones, Post-Accident
Annex Building, Elevation 100'-0" & 107'-2"

Security-Related Information — Withheld Under 10 CFR 2.390(d)
(See Part 9 of this COL Application)

(Note: This figure replaces DCD Figure 12.3-3 Sheet 11 of 16.)

VEGP DEP 18.8-1

Figure 12.3-203
Radiological Access Controls, Normal Operations/Shutdown
Annex Building, Elevation 100’-0” & 107’-2”

12.4 DOSE ASSESSMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

VEGP SUP 12.4-1 Add the following new subsections after DCD Subsection 12.4.1.8:

12.4.1.9 Radiation Exposure to Construction Workers

12.4.1.9.1 Site Layout

The physical location of VEGP Units 3 and 4 relative to Units 1 and 2 is depicted on Figure 1.1-202. As shown, Units 3 and 4 will be immediately west of Units 1 and 2. Construction activity will take place outside the Units 1 and 2 protected area, but inside the restricted area boundary.

12.4.1.9.2 Radiation Sources

During the construction of Units 3 and 4, the construction workers could be exposed to radiation sources from the routine operation of Units 1 and 2. Furthermore, Unit 4 construction workers could be exposed to radiation from Unit 3 operation.

12.4.1.9.2.1 Direct Radiation

The principal sources from Units 1 and 2 that contribute to direct radiation exposure at the construction site include the reactor buildings and the planned Independent Spent Fuel Storage Installation (ISFSI), which will be located east of Unit 1 (See Figure 1.1-202). In addition, workers constructing Unit 4 could be exposed to direct radiation from the Unit 3 reactor building.

12.4.1.9.2.2 Gaseous Effluents

Sources of gaseous releases for Units 1 and 2 are currently confined to the following paths: plant vents (Unit 1 and Unit 2), the condenser air ejector, the steam packing exhauster systems (Unit 1 and Unit 2), Radwaste Processing Facility and the DAW (Dry Active Waste Building). Waste gas decay tanks are batch released through the Unit 1 plant vent. The containment purges are released through their respective plant vents. (Reference 203)

The annual releases for the 2002 were reported as 26.3 Ci of fission and activation products, 0.0207 Ci of I-131, 1.67 x 10^{-5} Ci of particulates with half-lives greater than eight days, and 105 Ci of tritium (Reference 202). The annual releases for 2002 were selected because they resulted in the maximum exposure to the public among the years 2001-2004.

Unit 4 construction workers could also be exposed to radioactivity in gaseous effluents from Unit 3. DCD Table 11.3-3 presents the projected gaseous effluent releases for Unit 3.

12.4.1.9.2.3 Liquid Effluents

Effluents from the liquid waste disposal system result in small amounts of radioactivity in the Savannah River. The annual liquid radioactivity releases for 2001 were reported as 0.220 Ci of fission and activation products, 1,490 Ci tritium, and 0.000423 Ci of dissolved and entrained gases (Reference 201). The annual releases for 2001 were selected because they were reported as the maximum exposure to the public among the years 2001-2004.

Unit 4 construction workers could be exposed to radioactivity in liquid effluents from Unit 3, but that is unlikely given that drinking water is derived from sources other than the Savannah River. DCD Table 11.2-7 presents the projected liquid effluent releases for Unit 3. Applying the Units 1, 2, and 3 liquid effluent doses to Unit 4 construction workers is conservative in that it assumes these construction workers engage in the same activities that lead to the calculated liquid effluent doses (i.e., consuming fish and drinking surface water).

12.4.1.9.3 Measured and Calculated Dose Rates

The measured or calculated dose rates used to estimate worker doses are presented below.

12.4.1.9.3.1 Direct Radiation

Units 1 and 2 External Radiation Exposure

TLD data from 2003 is representative of annual results from Units 1 and 2, based on the completeness of the data set and having operated with a 95 percent plant capacity factor for that year. The average accumulated exposure from the six thermoluminescent dosimeters (TLDs) along the Units 1 and 2 Protected Area Fence closest to the construction site over a 365 day period is 115.9 mrem. The average TLD exposure from sixteen environmental locations surrounding the site over a 365 day period is 49.0 mrem. The measured radiation dose from the Protected Area Fence TLDs minus the Surrounding Environmental Site TLD's, is:

115.9 mrem per year - 49.0 mrem per year = 66.9 mrem per year

Independent Spent Fuel Storage Installation (ISFSI)

The dose to construction workers from the planned ISFSI is negligible for the Units 3 and 4 construction workforce.

Unit 3 Direct Radiation Exposure to Unit 4 Construction Workers

Conservatively assuming that the 66.9 mrem per year value presented above for Units 1 and 2 is attributable only to direct radiation from these units, and assuming this would be representative of the direct radiation dose from Unit 3 to Unit 4 construction workers gives a direct radiation dose to Unit 4 construction workers from Unit 3 operations of:

66.9 mrem per year / 2 units = 33.5 mrem per year (for one unit)

Summary of External Radiation

From all of the above sources discussed above, the highest direct radiation dose to construction workers will be during Unit 4 construction and is estimated to be 100.4 mrem per year (66.9 mrem from Units 1 and 2 + 33.5 mrem from Unit 3). The highest direct radiation exposure during Unit 3 construction is estimated to be 66.9 mrem per year (from Units 1 and 2). Therefore the Unit 4 construction workers doses would be bounding and are discussed in the remainder of this section.

12.4.1.9.3.2 Gaseous Effluents

Units 1 and 2

The XOQDOQ and GASPAR II codes were used to calculate the dose to Unit 4 workers from Units 1 and 2 gaseous effluents. The calculation is analogous to that for Units 3 and 4 as described in Subsection 11.3.3.4. Unit 4 construction workers would receive a total body radiation dose of 0.077 mrem per year and a maximum organ (lung) dose of 0.16 mrem per year from Units 1 and 2 normal radiological releases.

Unit 3 Gaseous Effluent Exposure to Unit 4 Construction Workers

Using the XOQDOQ and GASPAR II codes, as described in Subsection 11.3.3.4, Unit 4 construction workers would receive a total body radiation dose of 0.74 mrem per year and a maximum organ (skin) dose of 2.51 mrem per year from Unit 3 normal radiological releases.

12.4.1.9.3.3 Liquid Effluents

Units 1 and 2

The Annual Radioactive Effluent Release Report for 2001 (Reference 201) reports a total body dose of 0.0907 mrem and a critical organ dose (GI-LLI) of 0.153 mrem to the maximally exposed member of the public due to the release of liquid effluents from Units 1 and 2, calculated in accordance with Units 1 and 2 Offsite Dose Calculation Manual (Reference 204). SNC assumes this dose rate represents the rate for construction workers from Units 1 and 2 releases.

Unit 3 Liquid Effluent Exposure to Unit 4 Construction Workers

Using the LADTAP II code, as described in Subsection 11.2.3.5, the maximally exposed member of the public would receive a total body radiation dose of 0.017 mrem per year and a maximum organ (liver) dose of 0.021 mrem per year from normal Unit 3 liquid radiological releases.

12.4.1.9.4 Construction Worker Doses

Construction worker doses were conservatively estimated using the following information:

- The estimated maximum dose rate for each pathway
- An exposure time of 2000 hours per year
- All gaseous releases assumed at ground level
- A peak loading of 4,400 construction workers per year total for two AP1000 units

The estimated maximum annual dose for each pathway as well as the total dose is shown in Table 12.4-201.

12.4.1.9.4.1 Direct Radiation

Subsection 12.4.1.9.3.1 indicates an average annual direct radiation dose of 100.4 mrem based on TLD measurements. These TLD measurements and calculated doses reflect continuous exposures for long periods of time. The average measured dose rate of 100.4 mrem/yr is based on continuous exposure.

Adjusting for an exposure time of 2000 hours/year yields an annual worker whole body dose or total effective dose equivalent (TEDE) of 22.9 mrem.

12.4.1.9.4.2 Gaseous Effluents

The annual gaseous effluent doses to a Unit 4 construction worker after Unit 3 is operating (Subsection 12.4.1.9.3.2), which accounts for an exposure time of 2,000 hours per year, are 0.077 mrem for the total body, and 0.16 mrem for the critical organ (lung) from Units 1 and 2 gaseous effluent releases and 0.74 mrem for the total body, and 2.51 mrem (skin) for the critical organ from Unit 3 gaseous effluent releases. The total dose is 0.81 mrem total body and 2.60 mrem to the critical organ (skin).

12.4.1.9.4.3 Liquid Effluents

As the annual liquid effluent doses to the maximally exposed member of the public in Subsection 12.4.1.9.3 are based on continuous occupancy, they were adjusted for an exposure time of 2000 hr/yr. Although it is unlikely that the construction workers will be exposed to liquid effluent pathways, it is assumed that the liquid effluent dose rates to which the workers will be exposed are the same as those for the maximally exposed member of the public.

The resulting doses are 0.021 mrem for the total body and 0.035 mrem for the critical organ (GL-LLI) from Units 1 and 2 liquid effluent releases and 0.0038 mrem for the total body, and 0.0047 mrem for the critical organ (liver) from Unit 3 liquid effluent releases. The total annual dose is 0.025 mrem total body and 0.037 mrem to the critical organ (GI-LLI).

12.4.1.9.4.4 Total Doses

The annual doses from all three pathways are summarized in Table 12.4-201 and compared to the public dose criteria in the 10 CFR 20.1301 and 40 CFR 190 in Table 12.4-202 and Table 12.4-203, respectively. The unrestricted area dose rate in Table 12.4-202 was estimated from the annual TLD doses. Since the calculated doses (24.1 mrem per year and 0.012 mrem per hour) meet the public dose criteria of the 10 CFR 20.1301 and 40 CFR 190, the workers will not need to be classified as radiation workers. Table 12.4-204 provides documentation confirming that the doses also meet the design objectives of 10 CFR 50, Appendix I, for gaseous and liquid effluents.

The maximum annual collective dose to the AP1000 construction work force (4,400 workers) is estimated to be 106 person-rem. The calculated doses are based on available dose rate measurements and calculations. It is possible that these dose rates will increase in the future as site conditions change. However, the VEGP site will be continually monitored during the construction period and appropriate actions will be taken as necessary to ensure that the construction workers are protected from radiation.

12.4.1.9.4.5 Operating Unit Radiological Surveys

STD SUP 12.4-1 The operating unit conducts radiological surveys in the unrestricted and controlled area and radiological surveys for radioactive materials in effluents discharged to unrestricted and controlled areas in implementing 10 CFR 20.1302. These surveys demonstrate compliance with the dose limits of 10 CFR 20.1301 for construction workers.

Add the following new subsection after DCD Subsection 12.4.3

12.4.4 REFERENCES

201. Southern Nuclear Company, Vogtle Electric Generating Plant - Units 1 and 2, NRC Docket Nos. 50-424 and 50-425, Facility Operating License Nos. NPF-68 and NPF-81, Annual Radioactive Effluent Release Report for January 1, 2001 to December 31, 2001.

202. Southern Nuclear Company, Vogtle Electric Generating Plant - Units 1 and 2, NRC Docket Nos. 50-424 and 50-425, Facility Operating License Nos. NPF-68 and NPF-81, Annual Radioactive Effluent Release Report for January 1, 2002 to December 31, 2002.

203. Southern Nuclear Company, Vogtle Electric Generating Plant - Units 1 and 2, NRC Docket Nos. 50-424 and 50-425, Facility Operating License Nos. NPF-68 and NPF-81, Annual Radioactive Effluent Release Report for January 1, 2003 to December 31, 2003.

204. Southern Nuclear Company, Offsite Dose Calculation Manual for Southern Nuclear Operating Company Vogtle Electric Generating Plant, Version 22, June 25, 2004.

VEGP SUP 12.4-1

Table 12.4-201
Annual Construction Worker Doses

	Annual Dose (mrem)		
	Total Body	Critical Organ	Total Effective Dose Equivalent (TEDE)
Direct radiation	22.9	NA	22.9
Gaseous effluents	0.81	2.6 (skin)	1.16
Liquid effluents	0.025	0.037 (GI-LLI)	0.034
Total	23.8	2.6 (skin)	24.1

VEGP SUP 12.4-1

Table 12.4-202
Comparison with 10 CFR 20.1301 Criteria for Doses to Members of the Public

Criterion	Dose Limit	Estimated Dose (TEDE)
Annual dose (mrem)	100	24.1
Unrestricted area dose rate (mrem/hour)	2	0.012

VEGP SUP 12.4-1

Table 12.4-203
Comparison with 40 CFR 190 Criteria for Doses to Members of the Public

Organ	Annual Dose (mrem)	
	Limit	Estimated
Total body	25	23.8
Thyroid	75	1.4
Other organ	25	2.6 (skin)

VEGP SUP 12.4-1

Table 12.4-204
Comparison with 10 CFR 50,
Appendix I Criteria for Effluent Doses

	Annual Dose (mrem)	
	Limit	Estimated
Total body dose from liquid effluents	3	0.025
Organ dose from liquid effluents	10	0.037 (GI-LLI)
Total body dose from gaseous effluents	5	0.81
Organ dose from radioactive iodine and radioactive particulates in gaseous effluents	15	0.81 (thyroid)

12.5 HEALTH PHYSICS FACILITIES DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

12.5.2.2 Facilities

Revise the first sentence of DCD Subsection 12.5.2.2 to read:

VEGP DEP 18.8-1 The ALARA briefing room is located off the main corridor immediately beyond the main entry to the annex building.

12.5.4 CONTROLLING ACCESS AND STAY TIME

Add the following text to the end of DCD Subsection 12.5.4.

STD COL 12.3-1 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.

12.5.5 COMBINED LICENSE INFORMATION

STD COL 12.5-1 This COL Item is addressed in Appendix 12AA.

Add the following Appendix after Section 12.5 of the DCD.

APPENDIX 12AA RADIATION PROTECTION PROGRAM DESCRIPTION

STD COL 12.1-1
STD COL 12.3-1
STD COL 12.5-1

This appendix incorporates by reference NEI 07-03A, Generic FSAR Template Guidance for Radiation Protection Program Description. See Table 1.6-201. The numbering of NEI 07-03A is revised from 12.5# to 12AA.5# through the document, with the following revisions and additions as indicated by strikethroughs and underlines. Table 13.4-201 provides milestones for radiation protection program implementation.

Revise bullet number 3 of NEI 07-03A Section 12.5 as follows:

3. Prior to initial loading of fuel in the reactor, all of the radiation program functional areas described in Appendix 12AA~~Section12.5~~ will be fully implemented, with the exception of the organization, facilities, equipment, instrumentation, and procedures necessary for transferring, transporting or disposing of radioactive materials in accordance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71. In addition, the position of radiation protection manager (as described in ~~s~~Section 13.1 ~~12.5.2.3~~) will be filled and at least one (1) radiation protection technician for each operating shift, selected, trained, and qualified consistent with the guidance in Regulatory Guide 1.8, will be onsite and on duty when fuel is initially loaded in the reactor, and thereafter, whenever fuel is in the reactor.

Revise the first paragraph of NEI 07-03A Subsection 12.5.2 as follows:

Qualification and training criteria for site personnel are consistent with the guidance in Regulatory Guide 1.8 and are described in FSAR Chapter 13. Specific radiation protection responsibilities for key positions within the plant organization are described in Section 13.1 ~~below~~.

Subsections 12.5.2.1 through 12.5.2.5 of NEI 07-03A are not incorporated into Appendix 12AA.

Subsection 12.5.3.1 of NEI 07-03A is not incorporated into Appendix 12AA. Facilities are described in DCD Subsection 12.5.2.2.

Add the following text after the first paragraph of NEI 07-03A Subsection 12.5.3.3.

If circumstances arise in which NIOSH tested and certified respiratory equipment is not used, compliance with 10 CFR 20.1703(b) and 20.1705 is maintained.

The following headings (and associated material) in Subsection 12.5.4.2 of NEI 07-03A are described in DCD Subsection 12.5.3, and are therefore not incorporated into Appendix 12AA:

- Radwaste Handling
- Spent Fuel Handling
- Normal Operation
- Sampling

Add the following text after the second paragraph of NEI 07-03A Subsection 12.5.4.4.

STD COL 12.3-1 Table 12AA-201 identifies plant areas designated as Very High Radiation Areas (VHRAs), lists corresponding plant layout drawings showing the VHRA in DCD Section 12.3, specifies the condition under which the area is designated VHRA, identifies the primary source of the VHRA, and summarizes the frequency of access and reason for access. VHRAs are listed as Radiation Zone IX, which corresponds to a dose rate greater than 500 rad/hr.

In each of the VHRAs, with the exception of the Reactor Vessel Cavity and Delay-Bed / Guard-Bed Compartment, the primary radioactive source is transient (such as fuel passing through the transfer tube), removable (such as resin in the demineralizers), or can be relocated. When the primary source is removed, the dose rate in each of these areas will be less than Zone IX and, in effect, the area will no longer be a VHRA. With planning, the need for human entrance to a VHRA when the primary source is present can be largely or entirely avoided.

In addition to the access control requirements for high radiation areas, the following control measures are implemented to control access to very high radiation areas in which radiation levels could be encountered at 500 rads or more in one hour at one meter from a radiation source or any surface through which the radiation penetrates:

- Sign(s) conspicuously posted stating GRAVE DANGER, VERY HIGH RADIATION AREA.
- Area is locked. Each lock shall have a unique core. The keys shall be administratively controlled by the functional manager in charge of radiation protection as described in Section 13.1.
- Plant Manager’s (or designee) approval required for entry.

- Radiation Protection personnel shall accompany person(s) making the entry. Radiation Protection personnel shall assess the radiation exposure conditions at the time of the entry.

A verification walk down will be performed with the purpose of verifying barriers to the Very High Radiation Areas in the final design of the facility are consistent with Regulatory Guide 8.38 guidance as part of the implementation of the Radiation Protection and ALARA programs on the schedule identified in Table 13.4-201.

Revise the third paragraph of NEI 07-03A Subsection 12.5.4.7 as follows.

STD COL 12.1-1
STD COL 12.3-1
STD COL 12.5-1

As described in Sections 12.1, ~~12.5.1~~Appendix 12AA and ~~12.5.2~~ 13.1, management policy is established, and organizational responsibilities and authorities are assigned to implement an effective program for maintaining occupational radiation exposures ALARA. Procedures are established and implemented that are in accordance with 10 CFR 20.1101 and consistent with the guidance in Regulatory Guides 8.8 and 8.10. Examples of such procedures include the following:

Add the following text after the last bullet of NEI 07-03A Subsection 12.5.4.8.

STD COL 12.5-1

This subsection adopts NEI 08-08A (Reference 201), for a description of the operational and programmatic elements and controls that minimize contamination of the facility, site, and the environment, to meet the requirements of 10 CFR 20.1406.

Revise the first paragraph of Subsection 12.5.4.12 of NEI 07-03A to read:

STD COL 12.5-1

The radiation protection program and procedures are established, implemented, maintained, and reviewed consistent with the 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description described in Section 17.5.

Add the following Subsection to the information incorporated from NEI 07-03A.

12AA.5.4.14 Groundwater Monitoring Program

STD COL 12.3-3

A groundwater monitoring program beyond the normal radioactive effluent monitoring program is developed. If necessary to support this groundwater monitoring program, design features will be installed during the plant construction process. Areas of the site to be specifically considered in this groundwater monitoring program are (all directions based on plant standard):

- 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

This subsection adopts NEI 08-08A (Reference 201), for the Groundwater Monitoring Program description.

Add the following Subsection to the information incorporated from NEI 07-03A.

12AA.5.4.15 Record of Operational Events of Interest for Decommissioning

STD COL 12.3-4 This subsection adopts NEI 08-08A (Reference 201), for discussion of record keeping practices important to decomissioning.

Revise the REFERENCES section of NEI 07-03A, Reference 8, as follows:

8. Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."~~4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."~~

Add the following reference to NEI 07-03A REFERENCES.

201. NEI 08-08A, Generic FSAR Template Guidance for Life Cycle Minimization of Contamination, Revision 0, October 2009 (ML093220445).

STD COL 12.3-1

Table 12AA-201 (Sheet 1 of 2)
Very High Radiation Areas (VHRA)

Room Number	VHRA Location	DCD Figure 12.3-1, Sheet No.	Primary Source(s)	VHRA Conditional Notes	Frequency of Access to VHRA Areas While VHRA Conditions Exist
11105	Reactor Vessel Cavity	3, 4, 5	Neutron activation of the material in and around the cavity during reactor operations, such as the concrete shield walls and the reactor insulation	Note 1	None Required
12151	Spent Fuel Pool Cooling System / Liquid Radwaste System Demineralizer/ Filter room (Inside Wall)	3	Resin in vessels	Notes 6, 8	None Required
12153	Delay-Bed/ Guard-Bed Compartment	3	Activated carbon holding radioactive gases	Note 10	None Required
12371	Filter-Storage Area	6, 7	Spent filter cartridges	Notes 4, 6, 7	None required
12372	Resin Transfer Pump/Valve Room	6	Spent resin in lines	Note 6	None required
12373	Spent-Resin Tank Room	6	Spent resin in tanks	Note 6	None Required
12374	Waste Disposal Container Area	6	Spent resin in vault	Note 6	None Required
12463	Cask Loading Pit	6	Spent fuel	Notes 2, 6	None Required
12563	Spent Fuel Pit	5, 6	Spent fuel	Note 6	None Required
Fuel Transfer Areas					
12564	Fuel Transfer Tube	6	Fuel in transit	Notes 2, 5, 9	None Required
11205	Reactor Vessel Nozzle Area	5	Fuel in transit	Notes 2, 3, 9	None Required
11504	Refueling Cavity	6	Fuel in transit	Notes 2, 3, 9	None Required

STD COL 12.3-1

Table 12AA-201 (Sheet 2 of 2)
Very High Radiation Areas (VHRA)

Notes

1. VHRA during full power operation; less than 10 Rem/hr 24 hours after plant shutdown.
2. During underwater spent fuel transfer operations, this area can be as high as VHRA.
3. During underwater reactor internals transfers/ storage, this area can be as high as VHRA.
4. During spent resin waste disposal container transfer or loading, this area can be as high as VHRA. The contact dose rate of spent resin containers can be greater than 1000 Rem/hr.
5. Discussion about the Spent Fuel Transfer Canal and Tube Shielding is provided in DCD Subsection 12.3.2.2.9.
6. Source is transient, removable, or can be relocated.
7. VHRA when hatch is removed during spent resin container handling operation.
8. In the event that the room does need to be accessed for maintenance or other reasons, temporary shielding is put in place and the resin is removed from the vessels. These measures reduce exposure rates in the room, such that this room is no longer a VHRA. Remote handling is used for any tasks that require the opening of the access hatch in the ceiling of this room when media is present.
9. These areas have no planned reasons for entry and are only classified as VHRAs during periods of fuel movement. In the event that these rooms do need to be accessed to repair the Fuel-Transfer System, Fuel Transfer Tube Gate Valve, or other components, it is done during a non-fuel movement time. This keeps the dose received by the worker as low as reasonably achievable.
10. Inspection of the equipment in this room, when required, is done using remote viewing equipment. Two plugs between Room 12153 and 12155 contain instruments and the plugs are expected to be removed every 12 to 18 months for performance of maintenance. Administrative procedures are implemented to protect workers pursuant to Regulatory Guide 8.38.

CHAPTER 13

CONDUCT OF OPERATIONS

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CHAPTER 13
CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF APPLICANT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

VEGP DEP 1.1-1 DCD Subsection 13.1.1, Combined License Information, is renumbered in this FSAR section to 13.1.4.

VEGP COL 13.1-1 This section describes 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. Table 13.1-201, Generic Position/Site-Specific Position Cross-Reference, provides a cross-reference to identify the corresponding site-specific position titles.

13.1.1 MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

Southern Nuclear Operating Company, Inc. (SNC) has over 30 years of experience in the design, construction, and operation of nuclear generating plants. SNC, with its architectural engineering predecessor Southern Company Services, Inc., has designed, constructed, and currently operates six nuclear units at three sites: Edwin I. Hatch Nuclear Plant Units 1 and 2, Joseph M. Farley Nuclear Plant Units 1 and 2, and Vogtle Electric Generating Plant Units 1 and 2.

Vogtle 3 and 4 will be owned by Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and the City of Dalton, and operated by SNC.

13.1.1.1 Design, Construction, and Operating Responsibilities

The chief executive officer has overall responsibility for functions involving design, construction, and operation. Line responsibilities for those functions are assigned to the executive vice president of nuclear development (ND) for the design and construction of new nuclear plants and to the chief nuclear officer (CNO) for operation. At the appropriate time after construction, the CNO accepts responsibility for the additional units at the VEGP site from the executive vice president-ND and then maintains direct control of nuclear plant operation through the site vice president, operations support, engineering, and other direct reports.

The first priority and responsibility of each member of the nuclear staff throughout the life of the plant is nuclear safety. Decision-making for plant activities is performed in a conservative manner with expectations of this core value regularly communicated to appropriate personnel by management interface, training, and plant directives. Lines of authority, decision making, and communication are clearly and unambiguously established to enable the understanding of the various project members, including contractors, that utility management is in charge of and directs on the project. Key executive and corporate management positions, functions, and responsibilities are discussed in Subsection 13.1.1.3.1. The corporate organization is shown in Figure 13.1-201. The management and technical support organization for design, construction, and preoperational activities is addressed in Appendix 13AA.

13.1.1.2 Provisions for Technical Support Functions

Before beginning preoperational testing, the site vice president, the vice president - fleet operations support, and the vice president-engineering establish the organization of managers, functional managers, supervisors, and staff sufficient to perform required functions for support of safe plant operation. These functions include the following:

- Nuclear, mechanical, civil, structural, electrical, thermal-hydraulic, metallurgical and material, and instrumentation and controls engineering
- Safety review
- Quality assurance, audit and surveillance
- Plant chemistry
- Radiation protection and environmental support
- Fueling and refueling operations support
- Training
- Maintenance support
- Operations support
- Fire protection
- Emergency planning
- Outside contractual assistance

In the event that plant personnel are not qualified to deal with a specific problem, the services of qualified individuals from other functions within the company or an outside consultant are engaged. For example, major contractors, such as the reactor technology vendor or turbine generator manufacturer, provide technical support when equipment modifications or special maintenance problems are considered. Special studies, such as environmental monitoring, may be contracted to qualified consultants. Figure 13.1-202 illustrates the management and technical support organizations supporting operation of the plant. See Section 13.1.1.3.2 for description of responsibilities and authorities of management positions for organizations providing technical support. Table 13.1-201 shows the estimated number of positions required for each function.

Multiple layers of protection are provided to preserve unit integrity including organization. Organizationally, operators and other shift members are assigned to a specific unit, with the exception of some shared shift support supervisor and system operator positions. Physical separation of units helps to minimize wrong-unit activities. In addition, plant procedures and programs provide operating staff with methods to minimize human error including tagging programs, procedure adherence requirements, and training.

13.1.1.2.1 Engineering

The engineering department consists of system engineering, engineering programs, and engineering analysis. These groups are responsible for performing the site engineering activities as well as providing engineering expertise in other areas. Each of the engineering groups has a functional manager who reports to the engineering director.

The engineering department is responsible for:

- Support of plant operations in the engineering areas of mechanical, structural, electrical, thermal-hydraulic, metallurgy and materials, electronic, instrument and control, and fire protection. Priorities for support activities are established based on input from the plant manager with emphasis on issues affecting safe operation of the plant.

- Engineering programs.

- Support of procurement, chemical, and environmental analysis and maintenance activities in the plant as requested by the plant manager.

- Performance of design engineering of plant modifications.

- Human factors engineering design process

Reactor engineering, part of system engineering, provides technical assistance in the areas of core design, core operations, core thermal limits, and core thermal hydraulics.

Engineering work may be contracted to and performed by outside companies in accordance with the quality assurance (QA) program.

Engineering resources are shared between units. A single management organization oversees the engineering work associated with the plant units.

13.1.1.2.2 Safety Review

Review and audit activities are addressed in Chapter 17. Oversight of safety review of plant programs, procedures, and activities is performed by a plant review board and a corporate safety review board.

Personnel who perform safety review are shared between units.

13.1.1.2.3 Quality Assurance

Safety-related activities associated with the operation of the plant are governed by QA direction established in Chapter 17 of the FSAR and the QA Program Description (QAPD). The requirements and commitments contained in the QAPD apply to activities associated with structures, systems, and components that are safety-related and are mandatory and must be implemented, enforced, and adhered to by individuals and organizations. QA requirements are implemented through the use of approved procedures, policies, directives, instructions, or other documents that provide written guidance for the control of quality-related activities and provide for the development of documentation to provide objective evidence of compliance. The QA function includes:

- Maintaining the QAPD.
- Coordinating the development of audit schedules.
- Auditing, performing surveillance, and evaluating nuclear division suppliers.
- Supporting general QA indoctrination and training for the nuclear station personnel.

The QA organization is independent of the plant management line organization.

Quality control (QC) inspection/testing activities to support plant operation, maintenance, and outages are independent of the plant management line organization.

Personnel resources of the QA organization are shared between units. A single management organization oversees the QA group for the plant units.

13.1.1.2.4 Chemistry

A chemistry program is established to monitor and control the chemistry of various plant systems such that corrosion of components and piping is minimized and radiation from corrosion byproducts is kept to levels that allow operations and maintenance with radiation doses as low as reasonably achievable.

The chemistry manager is responsible to the plant manager for maintaining chemistry programs and for monitoring and maintaining the water chemistry of plant systems. The staff of the chemistry department consists of laboratory technicians, support personnel, and supervisors who report to the chemistry manager.

Personnel resources of the chemistry organization are shared between units. A single management organization oversees the chemistry group for the plant units.

13.1.1.2.5 Radiation Protection

A radiation protection (RP) program is established to protect the health and safety of the public and the personnel working at the plant. The RP program is described in Chapter 12 of the FSAR. The program includes:

- Respiratory protection
- Personnel dosimetry
- Bioassay
- Survey instrument calibration and maintenance
- Radioactive source control
- Effluents and environmental monitoring and assessment
- Radioactive waste shipping
- Radiation work permits
- Job coverage
- Radiation monitoring and surveys

The RP department is staffed by RP technicians, support personnel, and supervisors who report to the functional manager in charge of radiation protection. To provide sufficient organizational freedom from operating pressures, the manager in charge of RP reports directly to the plant manager.

Personnel resources of the RP organization are shared between units. A single management organization oversees the RP group for both units.

The terms radiation protection and health physics are used interchangeably in function descriptions.

13.1.1.2.6 Fueling and Refueling Support

The function of fueling and refueling is performed by a combination of personnel from various departments including operations, maintenance, radiation protection, engineering, and reactor technology vendor or other contractor staff. Initial fueling and refueling operations are a function of the outage support organization. The manager in charge of outage support is responsible for planning and scheduling outages and for refueling support and reports to the plant manager.

Personnel resources of the outage support organization are shared between units. A single management organization oversees outage support work associated with the station units.

13.1.1.2.7 Training and Development

The training department is responsible for analysis, design, development, implementation and evaluation of plant training programs in accordance with regulatory requirements, accreditation standards, and company policies. The objective of the training programs is to qualify operations, maintenance, RP, chemistry, and engineering personnel to operate and maintain the plant in a safe and efficient manner and in compliance with the license, technical specifications, and applicable regulations. In addition, the training department administers plant access (general employee) training, radworker training, security training, and emergency response training. The training manager reports independently of the line organization to provide for independence from operating pressures. Plant training programs are described in Section 13.2 of the FSAR.

Personnel resources of the training department are shared between units. A single management organization provides oversight of plant training activities.

13.1.1.2.8 Maintenance Support

In support of maintenance activities, planners, schedulers, and parts specialists prepare work packages, acquire proper parts, and develop procedures that provide for the successful completion of maintenance tasks. Maintenance tasks are integrated into the plant schedule for evaluation of operating or safe shutdown risk elements and to provide for efficient and safe performance. The maintenance manager reports to the plant manager.

Personnel of the maintenance support organization are shared between units. A single management organization oversees the function of maintenance support for the plant units.

13.1.1.2.9 Operations Support

The operations support function is provided under the direction of the operations manager. Operations support includes the following programs:

- Operations procedures
- Operations surveillances
- Equipment tagging

13.1.1.2.10 Fire Protection

VEGP COL 9.5-1 The plant is committed to maintaining a fire protection program as described in Section 9.5. The site vice president is responsible for the fire protection program. Assigning the responsibilities at that level provides the authority to obtain the resources and assistance necessary to meet fire protection program objectives, resolve conflicts, and delegate appropriate responsibility to fire protection staff. The relationship of the site vice president to other staff personnel with fire protection responsibilities is shown on Figure 13.1-202. Fire protection for the facility is organized and administered by the engineer in charge of fire protection. The engineer in charge of fire protection is trained and experienced in fire protection and nuclear plant safety or has available personnel who are trained and experienced in fire protection and nuclear plant safety.

The site vice president, through the engineer in charge of fire protection, is responsible for developing and implementing the fire protection program including developing fire protection procedures, site personnel and fire brigade training, and inspections of fire protection systems and functions. The engineer in charge of fire protection reports to the site vice president and coordinates operations-related fire protection program activities with the operations manager or his designee. Functional descriptions of position responsibilities are included in appropriate procedures. Plant personnel are responsible for adhering to the fire protection/ prevention requirements detailed in Section 9.5. The site vice president has the lead responsibility for the overall site fire protection during construction of new units.

Personnel resources of the fire protection organization are shared between units. A single management organization oversees the fire protection group for the plant units.

13.1.1.2.11 Emergency Organization

VEGP COL 13.1-1 The emergency organization is a matrixed organization composed of personnel who have the experience, training, knowledge, and ability necessary to implement actions to protect the public in the case of emergencies. Managers and plant personnel assigned positions in the emergency organization are responsible for supporting the emergency preparedness organization and emergency plan as required. The staff members of the emergency planning organization administrate and orchestrate drills and training to maintain qualification of plant staff members and develop procedures to guide and direct the emergency organization during an emergency. The staff is also responsible for maintaining emergency facilities and equipment. The functional manager in charge of emergency preparedness reports to the manager in charge of site support.

The site emergency plan organization is described in the Emergency Plan. Resources of the emergency planning group are shared between units. A single management organization oversees the emergency planning group for the plant units.

13.1.1.2.12 Outside Contractual Assistance

Contract assistance with vendors and suppliers of services not available from organizations established as part of utility staff is provided by the materials, purchasing, and contracts organization. Personnel in the materials, purchasing, and contracts organization perform the necessary functions to contract vendors of special services to perform tasks for which utility staff does not have the experience or equipment required. The functional manager in charge of materials, purchasing, and contracts reports to the department manager in charge of materials, purchasing, and contracts, and secondarily to the manager in charge of site support.

Resources of the materials, purchasing, and contracts organization are shared between units. A single management organization oversees the materials, purchasing, and contracts group for the plant units.

13.1.1.3 Organizational Arrangement

13.1.1.3.1 Executive Management Organization

The nuclear operations organization, under the supervision of the SNC president/CEO, has direct responsibility for the operation and maintenance of Southern Company's nuclear plants. Executive management establishes expectations such that a high level of quality, safety, and efficiency is achieved in aspects of plant operations and support activities through an effective management control system and an organization selected and trained to meet the above objectives. A high-level chart of the utility headquarters and engineering organization is illustrated in Figure 13.1-201. Executives and management with direct line of authority for activities associated with operation of the plant are shown in Figure 13.1-202. The structure of the nuclear operations organization is described in the following paragraphs. Portions of the SNC Fleet Operations Support, Engineering, Corporate Services, General Counsel and External Affairs, and Human Resources organizations are also described in the following paragraphs.

13.1.1.3.1.1 President and CEO

The president/CEO is responsible for all aspects of operation of Southern Company's nuclear plants including employment decisions. The president/CEO is also responsible for all technical and administrative support activities provided by SNC and nonaffiliated contractors. The president/CEO directs the executive vice president, the executive vice president nuclear development, the vice president and corporate counsel; the comptroller and treasurer; and the vice president-engineering in fulfillment of their responsibilities. The president/CEO reports to the Board of Directors with respect to all matters.

13.1.1.3.1.2 Executive Vice President — Nuclear Development

The executive vice president - nuclear development (EVP-ND) is responsible for new nuclear plant licensing, design, and construction as well as training and engineering for VEGP Units 3 and 4. The EVP-ND maintains control of nuclear plant construction through the management in charge of new nuclear plant licensing, project management, and the site vice president who reports to the EVP-ND until construction completion, following the guidelines of the Nuclear Development Quality Assurance Manual. The EVP-ND reports to the CEO.

13.1.1.3.1.3 Executive Vice President/Chief Nuclear Officer

The executive vice president (EVP) reports to the president/CEO. The EVP also serves as the CNO for SNC. The CNO assumes responsibility for the nuclear plant from the EVP-ND after construction of the plant. The CNO becomes responsible for overall plant nuclear safety and takes the measures needed to provide acceptable performance of the staff in operating, maintaining, and providing technical support to the plant. The EVP/CNO delegates authority and responsibility for the operation and support of the site through the site vice presidents, vice president of fleet operations support, vice president-engineering, and Fleet Oversight manager. It is the responsibility of the CNO to provide guidance and direction such that safety-related activities, including engineering, construction, operations, operations support, maintenance, and planning, are performed following the guidelines of the QA program for the operating units. The CNO has no ancillary responsibilities that might detract attention from nuclear safety matters. The EVP/CNO is also responsible for the safe, reliable, and efficient operation of the Edwin I. Hatch Nuclear Plant Units 1 and 2, the Joseph M. Farley Nuclear Plant Units 1 and 2, and the Vogtle Electric Generating Plant Units 1 and 2.

13.1.1.3.1.3.1 Site Vice President

The site vice president reports to the EVP-ND until construction completion. Following construction completion, the site vice president reports to the CNO. The site vice president is directly responsible for management and direction of activities associated with the efficient, safe, and reliable operation of the nuclear plant. The site vice president is assisted in management and technical support activities by the plant manager, the site support manager, and the site engineering director. The site vice president is responsible for the site fire protection program through the engineer in charge of fire protection. See Subsection 13.1.1.2.10.

13.1.1.3.1.3.2 Vice President-Fleet Operations Support

The vice president-fleet operations support reports to the executive vice president and is responsible for identifying and resolving fleet issues, and using trends, operating experience, and industry best practices to improve fleet performance. The vice president-fleet operations support directs the fleet operations manager, the fleet maintenance and work controls manager, the fleet refueling outage manager, the fleet chemistry and health physics manager, the nuclear fleet security and emergency preparedness manager, and the fleet training and performance improvement manager.

13.1.1.3.1.3.2.1 Manager of Fleet Security and Emergency Planning

The Nuclear Fleet Security and Emergency Planning (NFSEP) Manager reports to the Vice President–Fleet Operations Support and is responsible for management of the NFSEP organization and the overall coordination of fleet security activities and programs, the corporate emergency planning programs (including the common Emergency Operations Facility) and the Access Authorization program. The NFSEP Manager also has responsibility for site emergency response communication. The NFSEP organization is responsible for providing information and support concerning emergency plans and security to the Nuclear Development organization.

13.1.1.3.1.3.2.2 Manager of Fleet Training and Performance Improvement

VEGP COL 18.10-1 The manager in charge of nuclear training is responsible for supporting the development of training programs and providing fleet guidance to the function manager in charge of training at the site. The manager in charge of nuclear training reports to the vice president-fleet operations support.

13.1.1.3.1.3.3 Fleet Oversight Manager (Quality Assurance)

VEGP COL 13.1-1 The Fleet Oversight Manager reports to the Executive Vice President for the operations activities and to the Executive Vice President–Nuclear Development for the new reactor activities and is responsible for developing and maintaining the quality assurance programs, evaluating compliance to the programs and managing the Fleet Oversight organization resources.

13.1.1.3.1.4 Vice President-Engineering

The vice president-engineering reports directly to the CNO. This organization includes both project-specific and generic engineering support organizations for operating plants. Additionally, the vice president-engineering is responsible for nuclear fuel and nuclear licensing activities as well as the design support function.

13.1.1.3.1.5 Design Support Manager

The design support manager reports to the vice president-engineering and acts as the design authority for SNC plants. The design support manager directs the activities associated with configuration management and major plant modifications.

13.1.1.3.1.5.1 Nuclear Licensing Manager

The nuclear licensing manager reports to the vice president - engineering and directs operating plant-specific licensing supervisors, the probabilistic risk assessment (PRA) supervisor, and their staffs. The licensing supervisors provide matrixed accountability to the site vice president for their respective plants.

13.1.1.3.1.5.2 Nuclear Fuels Manager

The nuclear fuels manager is responsible for providing nuclear fuel and related business and technical support consistent with the operational needs of the plant. Activities include scheduling and procuring uranium concentrates, conversion, enrichment, and fabrication services. The department provides expertise and support for high-level waste disposal management. The nuclear fuels manager is assisted by an engineering staff and reports directly to the vice president-engineering.

13.1.1.3.1.6 Vice President and General Counsel

The vice president and general counsel reports to the president/CEO. This individual is responsible for the legal, compliance, and external affairs associated with the SNC plants. This individual is also responsible for external affairs activities which include governmental affairs, corporate communications, and environmental affairs. The vice president and general counsel is also the corporate secretary and directs the managing attorney/compliance manager, the environmental affairs manager, and the public affairs manager.

13.1.1.3.1.7 Comptroller and Treasurer

The comptroller and treasurer reports directly to the president/CEO and is responsible for the development, coordination, management, and communication of all financial matters, including budgeting and analysis, accounting and billing, and the maintenance of an effective system of internal controls.

VEGP COL 13.1-1 13.1.1.3.1.7.1 Supply Chain Manager

The supply chain general manager is responsible for providing direction and guidance to the site-located functional manager in charge of supply chain. The supply chain manager reports to the vice president-engineering.

13.1.1.3.2 Site Support Organization

As stated above, the site vice president is responsible for direct management of the plant, including industrial relations, planning, coordination, direction of operation, training, maintenance, refueling, and technical activities. Reporting to the site vice president in supportive roles are the site support manager and the site engineering director.

13.1.1.3.2.1 Site Support Manager

The site support manager is responsible for supporting the operating units. Reporting to the site support manager are the training manager, emergency preparedness supervisor, the site security manager, and the performance analysis supervisor.

13.1.1.3.2.2 Site Engineering Director

The site engineering director is the onsite lead position for engineering and reports to the site vice president. The site engineering director is responsible for engineering activities related to the operation or maintenance of the plant and design change implementation support activities. The site engineering director directs functional managers responsible for system engineering, design engineering, engineering programs, and engineering analysis.

13.1.1.4 Qualifications of Technical Support Personnel

VEGP COL 18.6-1 The qualifications of managers and supervisors of the technical support organization meet the qualification requirements in education and experience for those described in ANSI/ANS-3.1-1993 (Reference 201) as endorsed and amended by Regulatory Guide 1.8. The qualification and experience requirements of headquarters staff is established in corporate policy and procedure manuals.

13.1.2 OPERATING ORGANIZATION

VEGP COL 13.1-1

13.1.2.1 Plant Organization

As stated above, the site vice president is responsible for direct management of the plant, including industrial relations, planning, coordination, direction of operation, training, maintenance, refueling, and technical activities. Reporting to the site vice president, in addition to the supportive roles, is the plant manager.

The plant management, technical support, and plant operating organizations are shown in Figure 13.1-202. The on-shift operating organization is presented in Figure 13.1-203 which shows those positions requiring NRC licenses. Additional personnel are required to augment normal staff during outages.

Nuclear plant employees are responsible for reporting problems with plant equipment and facilities. They are required to identify and document equipment problems in accordance with the QA program. QA program requirements as they apply to the operating organization are described in Chapter 17. Administrative procedures or standing orders include:

- Establishment of a QA program for the operational phase.
- Preparation of procedures necessary to carry out an effective QA program. See Section 13.5 for description of the plant procedure program.
- A program for review and audit of activities affecting plant safety. See Section 17.5 for description of plant review and audit programs.
- Programs and procedures for rules of practice as described in Section 5.2 of N18.7-1976/ANS-3.2 (Reference 203).

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. Specifically, managers and supervisors are responsible for the following, as applicable to their position within the plant organization:

- Interface directly with RP staff to integrate RP measures into plant procedures and design documents and into the planning, scheduling, conduct, and assessment of operations and work.
- Notify RP personnel promptly when RP problems occur or are identified, take corrective actions, and resolve deficiencies associated with operations, procedures, systems, equipment, and work practices.
- Ensure department personnel receive RP training and periodic retraining, in accordance with 10 CFR 19 so that they are properly instructed and briefed for entry into restricted areas.

- Periodically observe and correct, as necessary, radiation worker practices.
- Support RP management in implementing the RP program.
- Maintain exposures to site personnel ALARA.

13.1.2.1.1 Plant Manager

The plant manager reports to the site vice president, is responsible for overall safe operation of the plant, and has control over those onsite activities necessary for safe operation and maintenance of the plant including the following:

- Operations
- Maintenance and modification
- Chemistry and radiochemistry
- Outage management

Additionally, the plant manager has overall responsibility for occupational and public radiation safety. RP responsibilities of the plant manager are consistent with the guidance in Regulatory Guide 8.8 and Regulatory Guide 8.10 including the following:

- Provide management RP policy throughout the plant organization.
- Provide an overall commitment to RP by the plant organization.
- Interact with and support the manager in charge of RP on implementation of the RP program.
- Support identification and implementation of cost-effective modifications to plant equipment, facilities, procedures, and processes to improve RP controls and reduce exposures.
- Establish plant goals and objectives for RP.
- Maintain exposures to site personnel ALARA.
- Support timely identification, analysis, and resolution of RP protection problems (e.g., through the plant corrective action program).
- Provide for training to site personnel on RP in accordance with 10 CFR 19.
- Establish an ALARA committee with delegated authority from the plant manager that includes, at a minimum, the managers in charge of operations, maintenance, engineering, and RP to help provide for effective

implementation of line organization responsibilities for maintaining worker doses ALARA.

As described in Subsection 13.1.2.1.2.1, the shift manager is the plant manager's direct representative for the conduct of operations. The succession of authority includes the authority to issue standing or special orders as required.

In the site vice president's absence, the plant manager assumes responsibility for the plant. The site vice president will designate in writing other qualified personnel to assume overall plant responsibility in his absence. The line of succession of authority and responsibility for overall operations in the event of unexpected events of a temporary nature is:

a. Operations manager

b. Site support manager

c. Maintenance manager

13.1.2.1.1.1 Engineering Support Manager

The Engineering Support Manager reports directly to the Engineering Director. The Engineering Support Manager is responsible for providing technical direction to other departments regarding the safe, efficient, and reliable operation of the systems and for reactor engineering. Along with the Engineering Support supervisors, the Engineering Support Manager provides direction and oversight for the system engineers, program engineers, reactor engineers, and Non-Destructive Examination (NDE) specialists/coordinator. The Engineering Support Manager also has the following responsibilities:

- Provide technical direction for equipment reliability review function.
- Act as the chairperson for the Equipment Reliability Board (ERB).
- Review and approve Temporary Modifications.

13.1.2.1.1.2 Systems Engineering Supervisor

Each Systems Engineering Supervisor reports directly to the Engineering Support Manager. The Engineering Supervisor provides oversight to systems or reactor engineers. The supervisor also has the following responsibilities:

- Provide technical direction to other departments regarding the safe, efficient, and reliable operation of systems and technical direction related to reactor engineering (system engineers and reactor engineers).
- Ensure training and qualification of personnel in accordance with applicable Engineering Training procedures.

- Complete assigned surveillance testing in accordance with frequencies in the Technical Specifications.
- Ensure proper design configuration control of structures, systems, and components.
- Function as acting Engineering Support Manager when necessary.

13.1.2.1.1.3 Programs Engineering Supervisor

The Programs Engineering Supervisor reports directly to the Engineering Support Manager and is responsible for duties defined under the Systems Engineering Supervisor in 13.1.2.1.1.2. In addition to those responsibilities, this programs supervisor also provides oversight to the programs engineers.

13.1.2.1.1.4 Design Engineering Manager

The Design Engineering Manager serves as the key design lead for the nuclear plant and functions as the primary interface between the Major Projects and Design Support departments in Corporate and the site's Change Control Board (CCB). Along with the Design Supervisor, the Design Engineering Manager facilitates design change package development and implementation. The Design Engineering Manager also has the following responsibilities:

- Provide technical oversight and approval of design products generated by the Design department.
- Ensure changes to plant design are technically adequate.
- Maintain administrative control of design calculations.
- Establish administrative controls for technical software.
- Interface with contracted Architect Engineers and other engineering firms providing design.
- Interface with Corporate Stress Analysis, Fire Protection and Environmental Qualification groups in Engineering Programs providing design.
- Ensure training and qualification of department personnel.

13.1.2.1.1.5 Performance Analysis Supervisor

The Performance Analysis Supervisor reports directly to the Site Support Manager and provides oversight to the Performance Analysis engineers and the Site Licensing Principal Engineer. The supervisor also ensures training and qualification of personnel is performed in accordance with applicable Training procedures. Activities related to this responsibility are:

- Monitor and disseminate Operating Experience to plant.
- Support licensing and regulatory compliance activities.
- Monitor effectiveness of Corrective Action Program.
- Maintain Commitment Tracking Program.
- Coordinate the Human Performance Improvement Program.
- Ensure adequate and timely periodic reporting to all appropriate agencies.
- Coordinate the development and updating of performance indicators.
- Coordinate plant Self-Assessment Program and benchmarking.

13.1.2.1.1.6 Fire Protection Engineer

The Fire Protection Engineer is responsible for the following:

- Fire protection program requirements, including consideration of potential hazards associated with postulated fires, knowledge of building layout, and system design.
- Design, maintenance, surveillance, and quality assurance of fire protection features (e.g., detection systems, suppression systems, barriers, dampers, doors, penetration seals and fire brigade equipment).
- Fire prevention activities (administrative controls and training).
- Fire brigade organization and training.
- Pre-fire planning.

In accordance with Regulatory Guide 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 six years of engineering experience, three of which were in a responsible position in charge of fire protection engineering work. The engineer in charge of fire protection is trained and experienced in fire protection and nuclear safety or has available personnel who are trained and experienced in fire protection and nuclear plant safety.

13.1.2.1.1.7 Health Physics Manager

The Health Physics Manager reports to the Plant Manager and serves as the "Radiation Protection Manager" for the facility referenced in Regulatory Guides 8.8 and 8.10. The Health Physics Manager has overall responsibility for the radiation protection program, has responsibility for plant activities involving radiological safety, has the authority to prevent unsafe work practices, and directs steps to prevent any unnecessary radiation exposure. The Health Physics Manager will ensure that Health Physics activities comply with the requirements of the plant operating license, Technical Specifications, approved fleet and plant procedures Security Plan, Emergency Plan, Quality Assurance Program, and applicable local, site, and federal regulations. The Health Physics Manager must meet or exceed the requirements of Regulatory Guide 1.8.

13.1.2.1.1.8 Health Physics Support Supervisor

The Health Physics Support Supervisor reports to the Health Physics Manager and is responsible for, but not limited to, the Health Physics support programs, like Dosimetry, ALARA, Rad Waste, Respiratory Protection, and fixed and portable radiological instrumentation. The Health Physics Support Supervisor may substitute for and perform the duties of the Health Physics Manager when designated and qualified under Regulatory Guide 1.8.

13.1.2.1.1.9 Health Physicist

The Health Physicist reports to the Health Physics Manager and is responsible for, but not limited to, monitoring Health Physics programs and indicators which include the ALARA program, radioactive waste management, shipping of radioactive material/waste, and outage preparedness. The Health Physicist tracks and evaluates performance indicators, supports industry benchmarking, implements special projects, and provides technical support. The Health Physicist may substitute for and perform the duties of the Health Physics Manager when designated and qualified.

13.1.2.1.1.10 Health Physics Foreman

Health Physics Foremen report to the Health Physics Manager and the Health Physics Support Supervisor and are directly responsible for, but not limited to, directing, scheduling, and coordinating the activities of the HP Technicians to support plant activities.

13.1.2.1.1.11 Health Physics Technician

Health Physics Technicians report to Health Physics Supervision. Their responsibilities include the following:

- Monitoring radiation controlled areas on a regularly scheduled basis using fixed and portable survey instruments to evaluate contamination, radiation fields and airborne radiation levels and measure dose rates for job coverage.

- Storing, issuing, and testing of respiratory protection equipment.
- Support calibrating fixed and portable radiation survey instruments.
- Providing health physics coverage of plant personnel to ensure safe radiological practices.
- Radiological evaluations of area decontamination activities.
- Prepare and authorize Radiation Work Permits.
- Provide support to the Dosimetry Program, as needed.

13.1.2.1.1.12 Maintenance Manager

The Maintenance Manager defines, communicates, and reinforces high standards for performance of maintenance activities and holds himself and others accountable for meeting those standards. The Maintenance Manager is responsible for:

- Establishing department goals and objectives.
- Ensuring that the roles and responsibilities of maintenance management personnel are communicated and reinforced.
- Setting standards for training of maintenance personnel to ensure that they have the necessary knowledge and skills.
- Maintaining the plant materiel condition.

13.1.2.1.1.13 Maintenance Superintendent

The Maintenance Superintendent reinforces standards of performance as described in the Maintenance Manager’s responsibilities and performs specific additional Maintenance Department roles, such as:

- Oversight for the maintenance support activities.
- Responsible for day-to-day administrative, training and qualification, and budget activities.
- Oversight of the Daily and Outage Planning Supervisors.
- Responsible for the Maintenance Procedure Program.

- Managing the Corrective Action Program for Maintenance.
- Responsible for the Predictive Maintenance Program.

13.1.2.1.1.14 Mechanical Maintenance, Electrical Maintenance, and Instrument and Controls Supervisors

Maintenance supervisors are responsible for:

- Coaching workers and reinforcing expectations and standards. During oversight of work activities, reinforcing positive behaviors, identifying when worker performance does not meet expectations, and addressing performance shortfalls.
- Ensuring that qualifications and proficiency of assigned personnel are consistent with the assigned work.
- Actively participating in training development and delivery, and helping identify performance gaps that can be addressed in training.
- Promoting craft ownership of equipment, processes, and programs to facilitate improvement.
- Executing the work schedule.

13.1.2.1.1.15 Fleet Oversight Supervisor and Staff

The Fleet Oversight Supervisor (FOS) is responsible to the Fleet Oversight Manager (FOM) for the direction of the assigned staff. The staff shall be located either at the plant site or the corporate office, and shall provide an independent review and evaluation of the implementation of the Quality Assurance Program (QAP).

In accordance with the QAP, the FOS shall have the authority from the FOM to stop or recommend stopping, through appropriate channels, unsatisfactory work which is not in compliance with the QAP.

Specific duties and the responsibilities of the FOS are:

- Developing and implementing procedures for audits, surveillances, procedure reviews, training and qualification and associated activities.

Specific duties and the responsibilities of the Fleet Oversight Staff are:

- Evaluating site and corporate activities for conformance to QAP requirements and procedures.

- Preparing a schedule of audits to be performed on site and corporate organizations and activities (line organization, contractors and/or suppliers, as appropriate).
- Performing planned and periodic audits of site and corporate organizations and activities (line organization, contractors and/or suppliers, as appropriate).
- Following up on audit findings until resolved and closed out.
- Preparing reports of audits, surveillances, reviews and other assigned activities and providing them to the FOM and appropriate line management.

13.1.2.1.2 Operations Department

Operations activities are conducted with safety of personnel, the public, and equipment as the overriding priority. The operations department is responsible for:

- Maintaining the plant in a safe condition at all times.
- Operating station equipment.
- Monitoring and surveillance of safety and nonsafety-related equipment.
- Fuel loading.
- Providing the nucleus of emergency and firefighting teams.

The operations department maintains sufficient licensed and senior licensed operators to staff the control room continuously using a crew rotation system. The operations department is under the authority of the operations manager, who, through his direct reports, directs the day-to-day operation of the plant.

Specific duties, functions, and responsibilities of key shift members are discussed in Subsection 13.1.2.1.2.1 and in plant administrative procedures and the technical specifications. The minimum shift manning requirements are shown in Table 13.1-202. Prior to fuel arriving on site, the necessary positions for unit operations will be filled.

Some resources of the operations organization are shared between units. Administrative and support personnel perform their duties on either unit. Additional operations staff is required to fill the on-shift staffing requirements of the additional units. To operate, or supervise the operation of more than one unit, an operator (senior reactor operator [SRO] or reactor operator [RO]) must hold an appropriate, active license for each unit. A single management organization oversees the operations group for the plant units. See Table 13.1-201 for estimated number of staff in the operations department.

The operations support section is staffed with sufficient personnel to provide support activities for the operating shifts and overall operations department. The following is an overview of the operations organization.

13.1.2.1.2.1 Operations Manager

The Operations Manager is responsible for the overall management of the Operations Department to ensure safe and efficient operation of the plant. The Operations Manager defines, communicates, and reinforces standards for performance. Reporting to him are the Outage Superintendent, Daily Superintendent, and Support Superintendent, and the Shift Managers. The Operations Manager:

- Ensures safe operation of the nuclear unit(s).
- Promotes a strong safety culture, with nuclear safety as an overriding priority.
- Establishes goals, objectives, and standards for operational activities.
- Provides direction in the training of operators.
- Monitors and assesses performance.
- Issues standing orders and special orders.

13.1.2.1.2.2 Operations Superintendents

The Outage Superintendent, Daily Superintendent, and Support Superintendent report to the Operations Manager and share the following duties and responsibilities:

- Provide direction to the Shift Managers for routine scheduling and coordination of Operations shift activities, including interfacing with other plant departments.
- Ensure plant operations are conducted per Technical Specifications, standing orders, the Offsite Dose Calculation Manual (ODCM) and approved procedures.
- Review and approve operating procedures, standing orders, and other special orders.
- May function as Operations Manager when designated.
- Supervise the preparation and review of plant operating procedures.

- Provide input to the Training Department for development and conduct of training and qualification of Operations Department personnel.
- Provide interface between Operations and other departments on administrative matters.

13.1.2.1.2.3 Shift Manager (SM)

The Shift Manager (SM) reports to the Operations Manager. The SM is the senior management representative on each shift and is responsible for the safe and efficient operation of the plant. The Shift Manager shall possess an SRO License. He has the following duties and responsibilities:

- Ensures plant operations are conducted in accordance with appropriate standing orders, the ODCM, unit operating procedures, and Technical Specifications.
- Maintains responsibility and oversight of activities that could affect core reactivity.
- Functions as Site Emergency Director when required.
- Has responsibility for the entire plant in the absence of the Site VP, Plant Manager, and Operations Manager. The SM has their authority, including issuing standing orders and other special orders, in their absence.
- Ensures the shift is properly manned, including the Fire Brigade.
- Provides leadership of crew in training and qualification programs.

13.1.2.1.2.4 Shift Supervisor (SS)

One Shift Supervisor (SS) is assigned to each operating unit on each shift. He is responsible for the safe and efficient operation of the assigned unit. Each SS shall possess an SRO License. The SS(s) report to the Shift Manager (SM) and have the following specific duties and responsibilities:

- Maintains responsibility and oversight of activities that could affect core reactivity.
- Ensures that plant operations are conducted per the Technical Specifications, ODCM, and approved procedures and standing orders.
- In charge of unit operation during startup, power operation, and shutdown.
- Supervise the Reactor Operators, Non-Licensed Operators, and Shift Support Supervisors to ensure proper performance of their assigned duties.

- Approves the removal of equipment and systems from service for maintenance, testing or operational activities.
- Authorizes maintenance and/or testing activities to be performed and ensures plant conditions are suitable for performing such activities. Maintains status of equipment and determines operability of equipment upon return to service.
- Ensures equipment clearances and tagging functions are performed.
- Administers the Operations Surveillance program.

13.1.2.1.2.5 Shift Support Supervisor (SSS)

The SSS reports to the Shift Manager (SM) and have the following duties and responsibilities:

- Monitors performance of operator rounds, system and equipment lineups, surveillances, and other routine shift activities.
- Ensures shift operations are conducted per Technical Specifications, ODCM, and approved procedures.
- Authorizes maintenance and/or testing activities to be performed, and ensures plant conditions are suitable.
- Issues equipment clearances.
- Responds during fire emergencies and acts as the Fire Brigade Leader in directing the fire fighting efforts of the Fire Brigade, as required.
- Coordinates the Fire Protection program for on shift operations.
- Assists with administration of the Operations Surveillance program.

13.1.2.1.2.6 Shift Technical Advisor (STA)

The shift technical advisor position meets the intent of NUREG-0660, as clarified by NUREG-0737, Item I.A.1.1. The STA position may be eliminated if the qualifications of the Shift Manager, Shift Supervisor, or SRO licensed Shift Support Supervisor meet the requirements of the STA position. Section 13.2 describes STA training, and Subsection 13.1.3 describes STA qualifications.

The STA position has the following responsibilities:

- Maintain independence from the normal operations shift as much as necessary to be able to make objective evaluations of plant operations and to advise or assist plant supervision in correcting conditions that may compromise safe operations.

- Serves in an advisory capacity to the SS. Upon entry into emergency operating procedures, the STA monitors and reports Critical Safety Functions to the SS.
- Investigates the causes of abnormal or unusual events, assesses adverse effects on plant operation, and reports any abnormality to the SS.

13.1.2.1.2.7 Reactor Operator (RO)

The Reactor Operators report to the SS.

Duties and responsibilities include:

- Operate the reactor and power plant safely.
- Perform reactivity changes.
- Maintain a broad awareness of activities in the Main Control Room and the plant.
- Monitor and control key parameters during normal operation.
- Perform shift operations and surveillance testing per approved procedures, standing orders, and Technical Specifications.

13.1.2.1.2.8 Non-Licensed Operator (NLO)

Non-Licensed Operators report to the respective unit's Shift Supervisor (SS) or Support Shift Supervisor (SSS).

Duties and responsibilities include:

- Monitor plant auxiliary equipment and/or systems outside the main control room.
- Operate systems in the field at the direction of the control room or SSS.
- Perform rounds to ensure proper operation of equipment.
- Remove equipment from service and execute clearance orders; restore equipment to service and remove clearances as directed by the SS or SSS.
- Respond to fire emergencies as a member of the Fire Brigade and perform fire fighting activities as directed by the Fire Brigade Leader.

- Perform assigned Emergency Response duties.

13.1.2.1.3 Conduct of Operations

VEGP COL 13.1-1 Plant operations are controlled and/or coordinated through the control room. Maintenance activities, surveillances, and removal from/return to service of structures, systems, and components affecting the operation of the plant may not commence without the approval of senior control room personnel. The rules of practice for control room activities, as described by administrative procedures, which are based on Regulatory Guide 1.114, address the following:

- Position/placement of operator at the controls workstation and the expected area of the control room where the majority of the time of the supervisor/manager in charge on shift should be spent.
- Definition and outline of “surveillance area” and requirement for continuous surveillance by the operator at the controls.
- Relief requirements for operator at the controls and the supervisor/ manager in charge on shift.

In accordance with 10 CFR 50.54:

- Reactivity controls may be manipulated only by licensed ROs and SROs except as allowed for training under 10 CFR 55.
- Apparatus and mechanisms other than controls that may affect reactivity or power level of the reactor shall be operated only with the consent of the operator at the controls or the manager/supervisor in charge on-shift.
- During operation of the facility in modes other than cold shutdown or refueling, an SRO shall be in the control room and a licensed RO or SRO shall be present at the controls.

13.1.2.1.4 Operating Shift Crews

Plant administrative procedures implement the required shift staffing. These procedures establish crews with sufficient qualified plant personnel to staff the operational shifts and be readily available in the event of an abnormal or emergency situation. The objective is to operate the plant with the required staff and to develop work schedules that minimize overtime for plant staff members who perform safety-related functions. Work hour limitations and shift staffing requirements defined by TMI Action Plan I.A.1.3 are retained in plant procedures. When overtime is necessary, the provisions in the technical specifications and the plant administrative procedures apply. Shift crew staffing plans may be modified during refueling outages to accommodate safe and efficient completion of outage work in accordance with the proceduralized work hour limitations.

The minimum composition of the operating shift crew is contingent on the unit operating status. Position titles, license requirements, and minimum shift staffing for various modes of operation are contained in technical specifications, administrative procedures, and Table 13.1-202, and illustrated in Figure 13.1-203.

13.1.2.1.5 Fire Brigade

The station is designed and the fire brigade organized to be self-sufficient with respect to firefighting activities. The fire brigade is organized to deal with fires and related emergencies that could occur. It consists of a fire brigade leader and a sufficient number of team members to be consistent with the equipment that must be put in service during a fire emergency. A sufficient number of trained and physically qualified fire brigade members are available on site during each shift. The fire brigade consists of at least five members on each shift. Members of the fire brigade are knowledgeable of building layout and system design. The assigned fire brigade members for any shift does not include the shift manager nor any other members of the minimum shift operating crew necessary for safe shutdown of the unit. It does not include any other personnel required for other essential functions during a fire emergency. Fire brigade members for a shift are designated in accordance with established procedures at the beginning of the shift.

13.1.3 QUALIFICATIONS OF NUCLEAR PLANT PERSONNEL

VEGP COL 18.6-1

13.1.3.1 Qualification Requirements

VEGP COL 13.1-1

Qualifications of managers, supervisors, operators, and technicians of the operating organization meet the qualification requirements in education and experience for those described in ANSI/ANS-3.1-1993 (Reference 201), as endorsed and amended by Regulatory Guide 1.8.

13.1.3.2 Qualifications of Plant Personnel

Résumés and/or other documentation of qualification and experience of initial appointees to appropriate management and supervisory positions are available for review by regulators upon request after position vacancies are filled.

13.1.4 COMBINED LICENSE INFORMATION ITEM

VEGP COL 13.1-1
VEGP DEP 1.1-1

This COL item is addressed in Subsections 13.1.1 through 13.1.3.

13.1.5 REFERENCES

201. American Nuclear Society, “American National Standard for Selection, Qualification, and Training of Personnel for Nuclear Power Plant,” ANSI/ANS -3.1-1993.

202. U.S. Nuclear Regulatory Commission, “Generic Letter 86-04, Policy Letter, Engineering Expertise on Shift.”

203. American Nuclear Society, “American National Standard for Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants,” N18.7-1976/ANS-3.2.

VEGP COL 18.6-1

VEGP COL 13.1-1

Table 13.1-201 (Sheet 1 of 4)
Generic Position/Site-Specific Position Cross-Reference

Nuclear Function	Function Position - ANSI/ANS-3.1-1993 section reference		Nuclear Plant Position (Site-Specific)	Expected Positions single unit	Expected additional positions 2nd unit
Executive management	chief executive officer		Chief Executive Officer (CEO)	1	-
	chief operating officer		N/A	N/A	-
	chief nuclear officer		Chief Nuclear Officer	1	-
	executive, nuclear generation and development		Executive Vice President, Nuclear Development	1	-
Nuclear support	executive, operations support		Fleet Operations Support Vice President	1	-
Plant management	executive		Site Vice President	1	-
	plant manager	4.2.1	Plant Manager	1	-
Engineering	executive		Engineering Vice President	1	-
	manager	4.2.4	Engineering Director	1	-
system engineering	functional manager	4.3.9	Engineering Supervisor	1	-
	system engineer		Engineer	23	10
design engineering	functional manager	4.3.9	Design Engineering Manager	1	-
	design engineer		Design Engineer	15	2
engineering programs	functional manager	4.3.9	Engineering Supervisor	1	-
	programs engineer		Engineer	10	2
safety and engineering analysis	functional manager	4.3.9	Performance Analysis Supervisor	1	-
	analysis engineer		Engineer	3	-
reactor engineering	functional manager	4.3.9	Engineering Supervisor	1	-
	reactor engineer		Engineer	3	1
engineering support	functional manager	4.3.9	Engineering Support Manager	1	-

VEGP COL 18.6-1
VEGP COL 13.1-1

Table 13.1-201 (Sheet 2 of 4)
Generic Position/Site-Specific Position Cross-Reference

Nuclear Function	Function Position - ANSI/ANS-3.1-1993 section reference		Nuclear Plant Position (Site-Specific)	Expected Positions single unit	Expected additional positions 2nd unit
Maintenance	manager	4.2.3	Maintenance Manager	1	-
instrumentation and control	functional manager	4.3.4	Team Leader, Instrumentation and Control	1	-
	supervisor	4.4.7	Assistant Team Leader, Instrumentation and Control	4	4
	technician	4.5.3.3	Instrumentation and Control Technician	30	15
mechanical	functional manager	4.3.6	Team Leader, Mechanical	1	-
	supervisor	4.4.9	Assistant Team Leader, Mechanical	2	-
	technician	4.5.7.2	Mechanic	40	15
electrical	functional manager	4.3.5	Team Leader, Electrical	1	-
	supervisor	4.4.8	Assistant Team Leader, Electrical	2	-
	technician	4.5.7.1	Electrician	20	10
support	functional manager	4.3	Maintenance Superintendent	1	-
Operations	manager	4.2.2	Operations Manager	1	-
operations, daily	functional manager	4.3.8	Operations Superintendent	1	-
operations, support	functional manager	4.3.8	Operations Superintendent	1	-
operations, (on-shift)	functional manager	4.4.1	Shift Manager	5	5
	supervisor	4.4.2	Shift Supervisor	5	5
	supervisor	4.4.2	Shift Support Supervisor	5	5
	licensed operator	4.5.1	Plant Operator	10	10
	non-licensed operator	4.5.2	System Operator	30	30
	shift technical advisor[a]	4.6.2	Shift Technical Advisor	5	5
Operations - rad waste	supervisor	4.4	N/A	-	-

VEGP COL 18.6-1

VEGP COL 13.1-1

Table 13.1-201 (Sheet 3 of 4)
Generic Position/Site-Specific Position Cross-Reference

Nuclear Function	Function Position - ANSI/ANS-3.1-1993 section reference		Nuclear Plant Position (Site-Specific)	Expected Positions single unit	Expected additional positions 2nd unit
operations, outage	functional manager	4.3.8	Operations Superintendent	1	-
Fire protection	supervisor	4.4	Fire Protection Engineer	1	-
Radiation protection	functional manager	4.3.3	Health Physics Manager	1	-
	supervisor	4.4.6	Health Physics Foreman	5	-
	technician	4.5.3.2	Health Physics Technician	18	9
	ALARA specialist		Health Physicist	3	1
Chemistry	functional manager	4.3.2	Chemistry Manager	1	-
	supervisor	4.4.5	Chemistry Foreman	5	-
	technician	4.5.3.1	Chemistry Technician	12	12
Nuclear safety assurance	manager	4.2	Nuclear Licensing Manager	1	-
licensing	functional manager	4.3	Site Nuclear Licensing Supervisor	1	-
	supervisor		N/A	-	-
	licensing engineer		Licensing Engineer	6	-
corrective action	functional manager	4.3	Fleet Training and Performance Improvement Manager	1	-
	corrective action specialist		Corrective Action Program Coordinator	2	1
emergency preparedness	functional manager	4.3	Emergency Preparedness Supervisor	1	-
	EP planner		EP Coordinator	2	-
Training	functional manager	4.3.1	Training Manager	1	-
	supervisor ops trng	4.4.4	Nuclear Operations Training Supervisor	1	1
	ops training instructor		Nuclear Operations Training Instructor	6	6
	supervisor tech staff/maint trng		Training Supervisor	1	-

VEGP COL 18.6-1

VEGP COL 13.1-1

Table 13.1-201 (Sheet 4 of 4)
Generic Position/Site-Specific Position Cross-Reference

Nuclear Function	Function Position - ANSI/ANS-3.1-1993 section reference		Nuclear Plant Position (Site-Specific)	Expected Positions single unit	Expected additional positions 2nd unit
	tech staff/maint instructors		Nuclear Plant Instructor	8	2
Purchasing, and contracts	functional manager	4.3	Supply Chain Manager	1	-
Security	functional manager	4.3	Security Manager	1	-
Planning and scheduling	functional manager	4.3	Outage and Scheduling Manager	1	-
	functional manager	4.3	N/A	-	-
	supervisor	4.4	Scheduling Supervisor	2	-
Quality assurance	functional manager	4.3.7	Manager, Fleet Oversight	1	-
	supervisor	4.4.13	Quality Assurance Supervisor	1	-
	QA auditor		QA Auditor	6	2
	supervisor	4.4.13	Quality Control Supervisor	2	-
	QC inspector		QC Inspector	4	2
Startup testing	supervisor	4.4.11	Startup Testing Supervisor	1	-
	startup test engineer		Startup Test Engineer	6	-
	supervisor	4.4.12	Preop Testing Supervisor	1	-
	preop test engineer		Preop Test Engineer	20	-

(a) The shift technical advisor position may be eliminated if the qualifications of the shift manager, shift supervisor, or SRO licensed shift support supervisor meet the requirements of the shift technical advisor position.

VEGP COL 13.1-1
VEGP COL 18.6-1

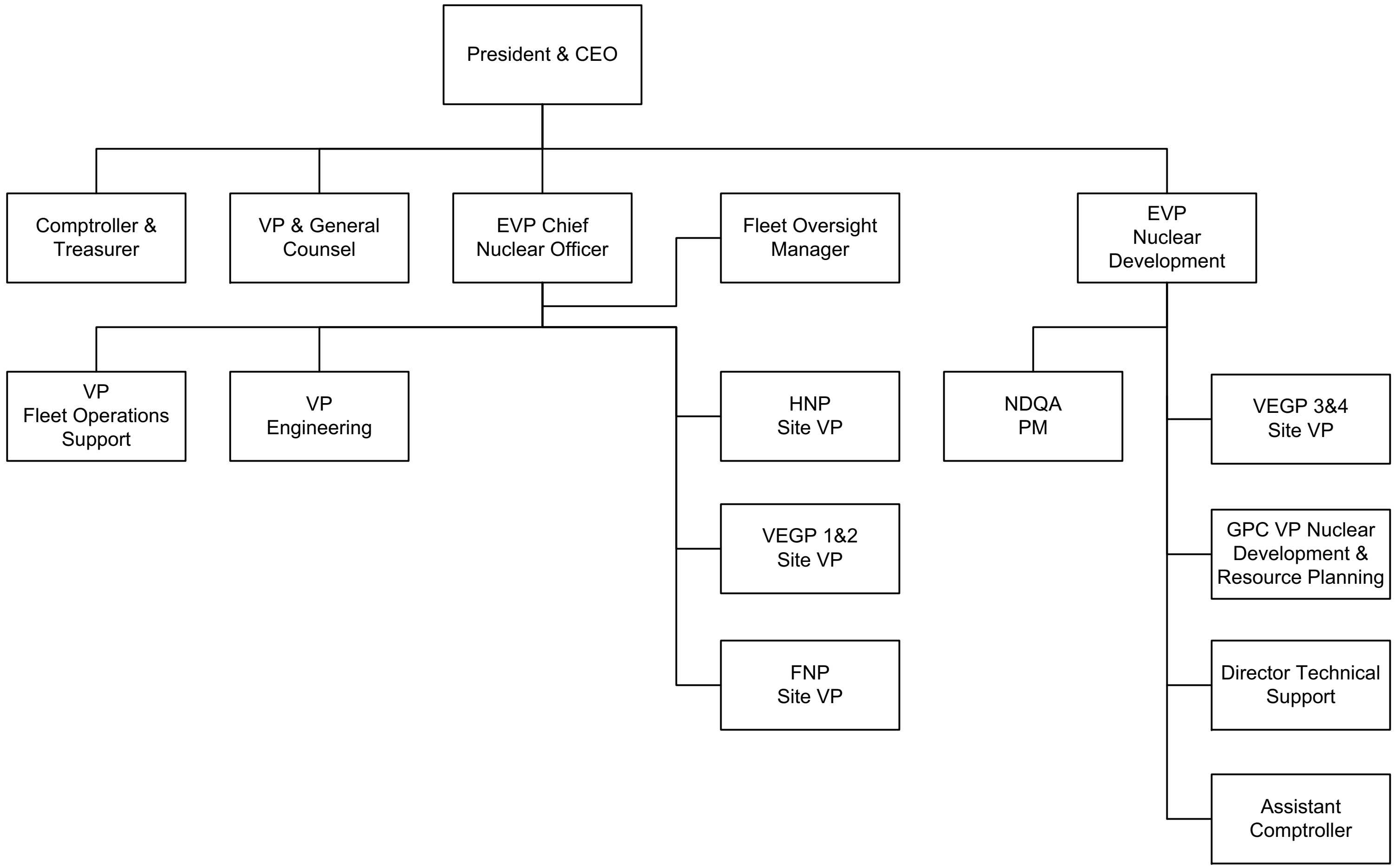
Table 13.1-202
Minimum On-Duty Operations Shift Organization for Two-Unit Plant

Units Operating	Two Units Two Control Rooms
All Units Shutdown	1 SM (SRO) 2 RO 3 NLO
One Unit Operating[(a)]	1 SM (SRO) 1 SRO 3 RO 3 NLO
Two Units Operating[(a)]	1 SM (SRO) 2 SRO 4 RO 4 NLO
SM – shift manager SRO – Licensed Senior Reactor Operator	RO – Licensed Reactor Operator NLO – non-licensed operator

a) Operating modes other than cold shutdown or refueling.

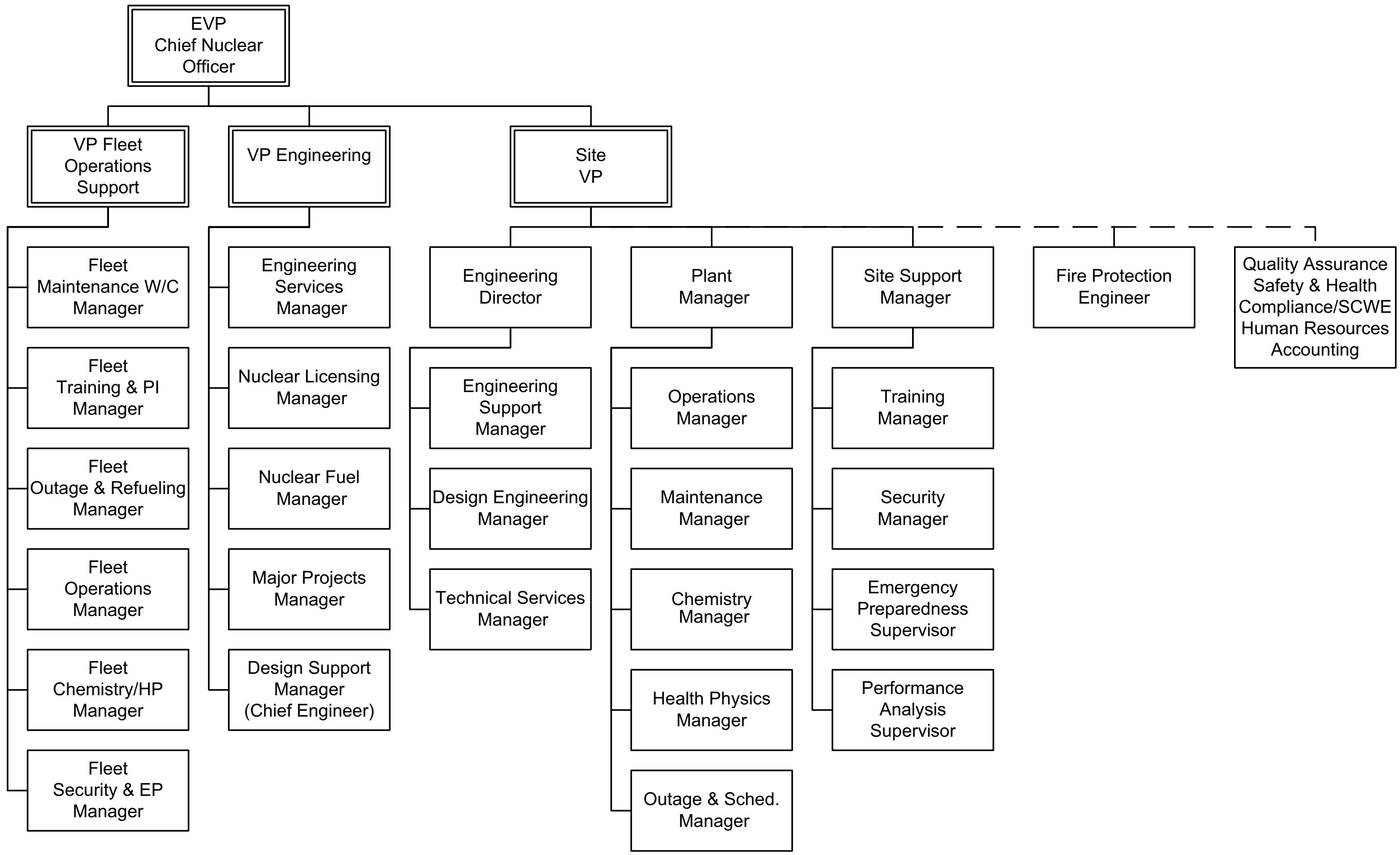
Notes:

1. In addition, one Shift Technical Advisor (STA) is assigned per shift during plant operation. A shift manager or another SRO on shift, who meets the qualifications for the combined Senior Reactor Operator/Shift Technical Advisor position, as specified for option 1 of Generic Letter 86-04, (Reference 202) the commission's policy statement on engineering expertise on shift, may also serve as the STA. If this option is used for a shift, then the separate STA position may be eliminated for that shift.
2. In addition to the minimum shift organization above, during refueling a licensed senior reactor operator or senior reactor operator limited (fuel handling only) is required to directly supervise any core alteration activity.
3. A shift manager/supervisor (SRO licensed for each unit that is fueled), shall be on site at all times when at least one unit is loaded with fuel.
4. A radiation protection technician shall be on site at all times when there is fuel in a reactor.
5. A chemistry technician shall be on site during plant operation in modes other than cold shutdown or refueling.
6. To operate, or supervise the operation of more than one unit, an operator (SRO or RO) must hold an appropriate, current license for each unit.



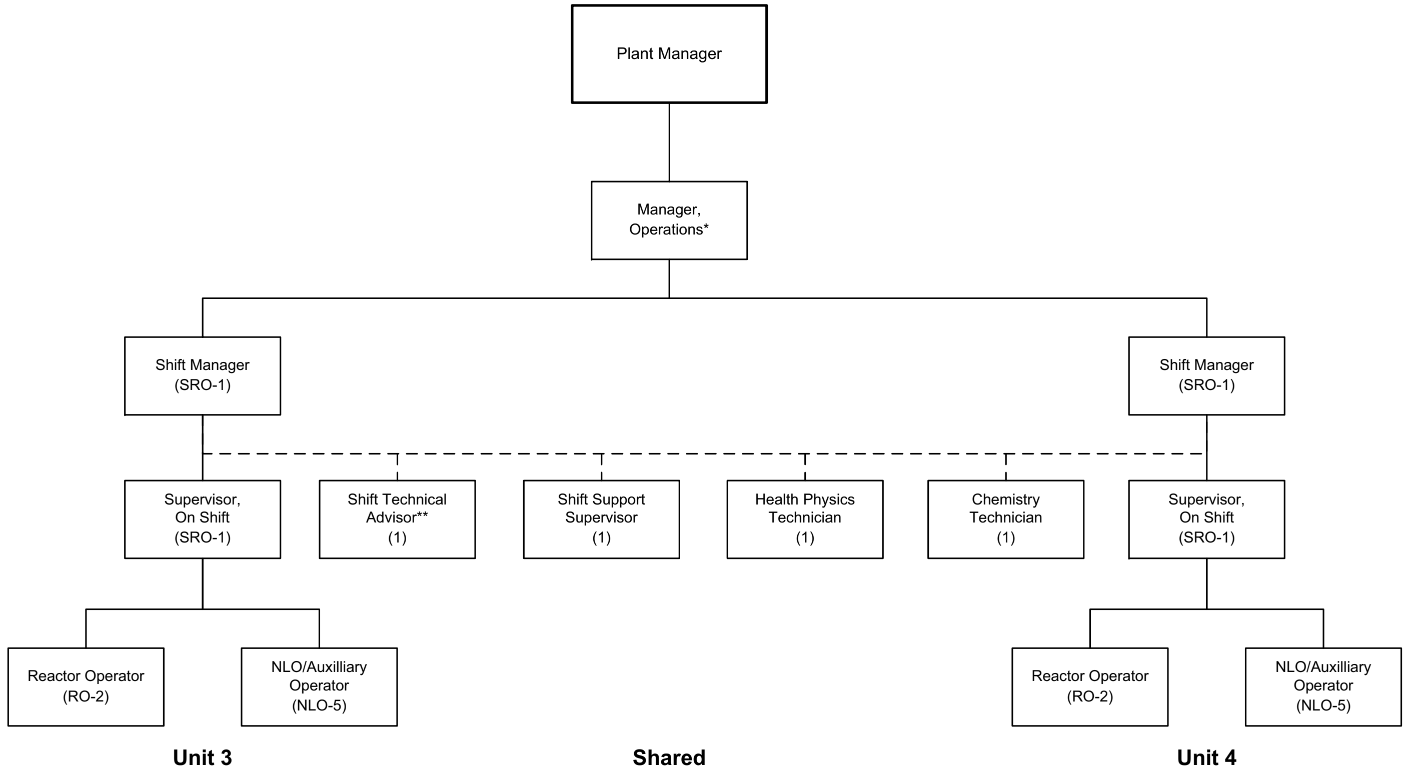
VEGP COL 13.1-1

Figure 13.1-201
Corporate and Engineering Organization



VEGP COL 13.1-1

Figure 13.1-202
Plant Management Organization



*The Operations Manager or Assistant Operations Manager shall hold an SRO license.
**May be met by onshift combined SRO/STA.

SRO - licensed senior reactor operator
RO - licensed reactor operator
NLO - non-licensed operator
Shift Manning - 5 shifts (minimum)
(No.) - indicate number of positions per shift

VEGP COL 13.1-1

Figure 13.1-203
Shift Operations

13.2 TRAINING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD COL 13.2-1 This section incorporates by reference NEI 06-13A, Template for an Industry Training Program Description. See Table 1.6-201.

Table 13.4-201 provides milestones for training implementation.

STD COL 18.10-1 Operators involved in the Human Factors Engineering Verification and Validation (V&V) Program receive additional training specific to the task of performing V&V. A systematic approach to training is incorporated in developing this training program along with input from WCAP-14655, Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel (Reference 201).

13.2.1 COMBINED LICENSE INFORMATION ITEM

STD COL 13.2-1 This COL Item is addressed in Section 13.2.

13.2.2 REFERENCES

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201. Westinghouse, "Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel," WCAP-14655, Revision 1, August 1996.

13.3 EMERGENCY PLANNING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

This section of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

Add the following text after the last paragraph of ESPA SSAR Section 13.3.

STD COL 13.3-1 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 dc equipment rooms are established in operating procedures.

STD COL 13.3-2 The emergency plan describes the plans for coping with emergency situations, including communications interfaces and staffing of the emergency operations facility.

STD SUP 13.3-1 Table 13.4-201 provides milestones for emergency planning implementation.

Subsection 13.3.1 of the DCD is renumbered as 13.3.6 to allow for sequential numbering and incorporation of the ESPA SSAR Emergency Planning information in the FSAR.

VEGP DEP 1.1-1

13.3.6 COMBINED LICENSE INFORMATION ITEM

STD COL 13.3-1 This COL Item is addressed in Section 13.3.

STD COL 13.3-2 This COL Item is addressed in Section 13.3 and in the Emergency Plan.

Add the following sections after Subsection 13.3.6.

VEGP SUP 13.3-2

13.3.7 NEW OR ADDITIONAL INFORMATION

In accordance with 10CFR 52.79(b)(4) and 10CFR 50.54(q), the following new and additional emergency planning information would materially change the bases for compliance with emergency planning requirements.

The VEGP Early Site Permit application Emergency Plan proposed a set of emergency action levels (EAL) based on a proposed industry guideline (NEI 07-01). Because of uncompleted design details related to the AP1000, certain details of site specific EALs based on NEI 07-01 cannot be completed until after the Combined License is scheduled to be issued. Consequently, the proposed VEGP EAL scheme is being removed from the VEGP Emergency Plan. Details of the changes are listed in VEGP COLA Part 5. A proposed License Condition 4, Emergency Planning Actions, has been added to Part 10 Proposed License Conditions (Including ITAAC). The license condition commits VEGP to submitting a complete set of EALs based on a NRC endorsed version of NEI 07-01 to the NRC at least 180 days prior to scheduled fuel load.

Subsequent to the VEGP Early Site Permit application design details related to the building in which the TSC will be located were refined. The design change resulted in the TSC being relocated from a proposed administration building to a proposed Communication Support Center. The change also resulted in moving the TSC approximately 150 feet east of the location identified in the Early Site Permit Emergency Plan. The changes are reflected in revised drawing Figure ii, Vogtle Electric Generating Plant Site Plan, and Section H of the Emergency Plan.

The above changes would constitute a reduction in effectiveness in regards to 10 CFR 10 CFR 50.54(q). However, the planning standards and regulations in 10 CFR 50.47 and 10 CFR 50 Appendix E continue to be met. Consequently, these changes require prior NRC approval in accordance with 10 CFR 50.54(q). Approval of these changes is requested as part of this COLA.

13.3.8 ESP PERMIT CONDITIONS

VEGP ESP PC 2 thru 7 SNC will revise and submit the VEGP Units 3 and 4 Emergency Action Levels (EALs) in accordance with License Condition No. 4 identified in Part 10, Proposed License Conditions (Including ITAAC).

VEGP ESP PC 8 Location of the Technical Support Center (TSC) is described in the Emergency Plan.

13.4 OPERATIONAL PROGRAMS

This section of the referenced DCD is incorporated by reference with the following departures and /or supplements.

STD COL 13.4-1 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).

13.4.1 COMBINED LICENSE INFORMATION ITEM

STD COL 13.4-1 This COL Item is addressed in Section 13.4.

13.4.2 REFERENCES

201. ASME Boiler and Pressure Vessel Code (B&PVC), “Section XI - Rules for Inservice Inspection of Nuclear Power Plant Components.”

202. ASME “OM Code for the Operation and Maintenance of Nuclear Power Plants.”

STD COL 13.4-1

Table 13.4-201 (Sheet 1 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
1.	Inservice Inspection Program	10 CFR 50.55a(g)	5.2.4, 5.4.2.5, 6.6	Prior to Commercial service	10 CFR 50.55a(g), ASME XI IWA-2430(b) (Reference 201)
2.	Inservice Testing Program	10 CFR 50.55a(f); 10 CFR Part 50, Appendix A	3.9.6, 5.2.4	After generator online on nuclear heat[(a)]	10 CFR 50.55a(f), ASME OM Code (Reference 202)
3.	Environmental Qualification Program	10 CFR 50.49(a)	3.11	Prior to initial fuel load	License Condition
4.	Preservice Inspection Program	10 CFR 50.55a(g)	5.2.4, 5.4.2.5, 6.6	Completion prior to initial plant start-up	10 CFR 50.55a(g); ASME XI IWB-2200(a) (Reference 201)
5.	Reactor Vessel Material Surveillance Program	10 CFR 50.60; 10 CFR 50.61; 10 CFR Part 50, Appendix H	5.3.2.6	Prior to initial criticality	License Condition
6.	Preservice Testing Program	10 CFR 50.55a(f)	3.9.6	Prior to initial fuel load	License Condition

Table 13.4-201 (Sheet 2 of 9)
Operational Programs Required by NRC Regulations

STD COL 13.4-1

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
7.	Containment Leakage Rate Testing Program	10 CFR 50.54(o); 10 CFR 50, Appendix A (GDC 52); 10 CFR 50, Appendix J	6.2.5.1	Prior to initial fuel load	License Condition
8.	Fire Protection Program	10 CFR 50.48	9.5.1.8	Prior to receipt of fuel onsite Prior to initial fuel load	License Condition
	(portions applicable to radioactive material)	10 CFR 30.32 10 CFR 40.31 10 CFR 70.22		Prior to initial receipt of byproduct, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18)	10 CFR 30.32(a) 10 CFR 40.31(a) 10 CFR 70.22(a)
9.	Process and Effluent Monitoring and Sampling Program:				
	Radiological Effluent Technical Specifications/ Standard Radiological Effluent Controls	10 CFR 20.1301 and 20.1302; 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR 50, Appendix I, Section II and IV	11.5	Prior to initial fuel load	License Condition
	Offsite Dose Calculation Manual	Same as above	11.5	Prior to initial fuel load	License Condition

STD COL 13.4-1

Table 13.4-201 (Sheet 3 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
	Radiological Environmental Monitoring Program	Same as above	11.5	Prior to initial fuel load	License Condition
	Process Control Program	Same as above	11.4	Prior to initial fuel load	License Condition
10.	Radiation Protection Program (including ALARA principle)	10 CFR 20.1101 10 CFR 20.1406	12.1 12.5		License Condition
	• Radioactive Source Control (assignment of RP Supervisor) • Assignment of RP Supervisor • Minimization of Contamination			1. Prior to initial receipt of by-product, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18)	
	• Personnel Dosimetry • Radiation Monitoring and Surveys • Radiation Work Permits			2. Prior to receipt of fuel onsite	
	• Assignment of RP Manager • Respiratory Protection • Bioassay			3. Prior to initial fuel load	

STD COL 13.4-1

Table 13.4-201 (Sheet 4 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
	• Effluents and Environmental Monitoring and Assessment				
	• Job Coverage				
	• Radioactive Waste Shipping			4. Prior to first shipment of radioactive waste	
11.	Non Licensed Plant Staff Training Program	10 CFR 50.120	13.2	18 months prior to scheduled date of initial fuel load	10 CFR 50.120(b)
	(portions applicable to radioactive material)	10 CFR 30.32 10 CFR 40.31 10 CFR 70.22		Prior to initial receipt of byproduct, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18)	10 CFR 30.32(a) 10 CFR 40.31(a) 10 CFR 70.22(a)
12.	Reactor Operator Training Program	10 CFR 55.13; 10 CFR 55.31; 10 CFR 55.41; 10 CFR 55.43; 10 CFR 55.45	13.2	18 months prior to scheduled date of initial fuel load	License Condition
13.	Reactor Operator Requalification Program	10 CFR 50.34(b); 10 CFR 50.54(i); 10 CFR 55.59	13.2	Within 3 months after the date the Commission makes the finding under 10 CFR 52.103(g)	10 CFR 50.54 (i-1)

STD COL 13.4-1

Table 13.4-201 (Sheet 5 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
14.	Emergency Planning	10 CFR 50.47; 10 CFR 50, Appendix E	13.3	Full participation exercise conducted within 2 years of scheduled date for initial loading of fuel.	10 CFR Part 50, Appendix E, Section IV.F.2.a(ii)
				Onsite exercise conducted within 1 year before the schedule date for initial loading of fuel	10 CFR Part 50, Appendix E, Section IV.F.2.a(ii)
				Applicant's detailed implementing procedures for its emergency plan submitted at least 180 days prior to scheduled date for initial loading of fuel	10 CFR Part 50, Appendix E, Section V
15.	Security Program:				
	Physical Protection Program (applicable to protection of special nuclear material prior to the protected area being declared operational)	10 CFR 73.1 10 CFR 73.67	13.5.2.2.8 13.6	Prior to initial receipt of special nuclear material	10 CFR 73.1(a) 10 CFR 73.67
	Physical Security Program	10 CFR 73.55(b); 10 CFR 73.55(c)(3); 10 CFR 73.56; 10 CFR 73.57;	13.6	Prior to receipt of fuel onsite (protected area)	10 CFR 73.55(a)(4)

STD COL 13.4-1

Table 13.4-201 (Sheet 6 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
	Safeguards Contingency Program	10 CFR 73.55(c)(5); 10 CFR 73.55(k); 10 CFR Part 73, Appendix C	13.6	Prior to receipt of fuel onsite (protected area)	10 CFR 73.55(a)(4)
	Training and Qualification Program	10 CFR 73.55(c)(4); 10 CFR 73.55(d)(3); 10 CFR Part 73, Appendix B	13.6	Prior to receipt of fuel onsite (protected area)	10 CFR 73.55(a)(4)
16.	Quality Assurance Program – Operation	10 CFR 50.54(a); 10 CFR Part 50, Appendix A (GDC 1); 10 CFR Part 50, Appendix B	17.5	COL issuance	10 CFR 50.54(a)(1)
17.	Maintenance Rule	10 CFR 50.65	17.6	Prior to fuel load authorization per 10 CFR 52.103(g)	10 CFR 50.65(a)(1)
18.	Motor-Operated Valve Testing	10 CFR 50.55a(b)(3)(ii)	3.9.6.2.2	Prior to initial fuel load	License Condition

STD COL 13.4-1

Table 13.4-201 (Sheet 7 of 9)
Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
19.	Initial Test Program	10 CFR 50.34; 10 CFR 52.79(a)(28)	14.2	Prior to the first construction test being conducted for the Construction Test Program Prior to the first preoperational test for the Preoperational Test Program Prior to initial fuel load for the Startup Test Program	License Condition
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

Table 13.4-201 (Sheet 8 of 9)
Operational Programs Required by NRC Regulations

STD COL 13.4-1

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
	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

Table 13.4-201 (Sheet 9 of 9)
Operational Programs Required by NRC Regulations

STD COL 13.4-1

Item	Program Title	Program Source (Required by)	FSAR Section	Implementation Milestone	Implementation Requirement
	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
21.	Cyber Security Program	10 CFR 73.54(b); 10 CFR 73.55(b)(8); 10 CFR 73.55(c)(6)	13.6	Prior to receipt of fuel onsite (protected area)	10 CFR 73.55(a)(4)
22.	SNM Material Control and Accounting Program	10 CFR 74, Subpart B (§§ 74.11 – 74.19, excl. § 74.17)	13.5.2.2.9	Prior to receipt of special nuclear material	License Condition

(a) Inservice Testing Program will be fully implemented by generator on line on nuclear heat. Appropriate portions of the program are implemented as necessary to support the system operability requirements of the technical specifications.

13.5 PLANT PROCEDURES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

VEGP DEP 1.1-1 DCD Subsection 13.5.1, Combined License Information, is renumbered in this FSAR section to 13.5.3.

STD COL 13.5-1 This section of the FSAR describes the administrative and other procedures which are not described in the DCD that the operating organization (plant staff) uses to conduct the routine operating, abnormal, and emergency activities in a safe manner.

The Quality Assurance Program Description (QAPD), as discussed in Section 17.5, describes procedural document control, record retention, adherence, assignment of responsibilities, and changes.

Procedures are identified in this section by topic, type, or classification in lieu of the specific title and represent general areas of procedural coverage.

Procedures are issued prior to fuel load to allow sufficient time for plant staff familiarization and to develop operator licensing examinations.

The format and content of procedures are controlled by the applicable AP1000 Writer's Guideline.

Each procedure is sufficiently detailed for an individual to perform the required function without direct supervision, but does not provide a complete description of the system or plant process. The level of detail contained in the procedure is commensurate with the qualifications of the individual normally performing the function.

Procedures are developed consistent with guidance described in DCD Section 18.9, "Procedure Development" and with input from the human factors engineering process and evaluations.

13.5.1 ADMINISTRATIVE PROCEDURES

This section describes administrative procedures that provide administrative control over activities that are important to safety for the operation of the facility.

Procedures outline the essential elements of the administrative programs and controls as described in ANSI/ANS 3.2-1988 (Reference 201) and in Section 17.5. These procedures are organized such that the program elements are prescribed in documents normally referred to as administrative procedures. Regulatory and industry guidance for the appropriate format, content and typical activities delineated in written procedures is implemented as appropriate.

Administrative procedures contain adequate programmatic controls to provide effective interface between organizational elements. This includes contractor and owner organizations providing support to the station operating organization.

A Writer's Guideline promotes the standardization and application of human factors engineering principles to procedures. The Writer's Guideline establishes the process for developing procedures that are complete, accurate, consistent, and easy to understand and follow. The Writer's Guideline provides objective criteria so that procedures are consistent in organization, style, and content. The Writer's Guideline includes criteria for procedure content and format including the writing of action steps and the specification of acceptable acronym lists and acceptable terms to be used.

Procedure maintenance and control of procedure updates are performed in accordance with the QAPD, as discussed in Section 17.5.

The administrative programs and associated procedures developed in the pre-COL phase are described in Table 13.5-201 (for future designation as historical information).

The plant administrative procedures provide procedural instructions for the following:

- Procedures review and approval.
- Equipment control procedures - These procedures provide for control of equipment, as necessary, to maintain personnel and reactor safety, and to avoid unauthorized operation of equipment.
- Control of maintenance and modifications.
- Crane Operation Procedures - Crane operators who operate cranes over fuel pools are qualified and conduct themselves in accordance with ANSI B30.2 (Chapter 2-3), "Overhead and Gantry Cranes" (Reference 202).
- Temporary changes to procedures.
- Temporary procedure issuance and control.
- Special orders of a temporary or self-canceling nature.
- Standing orders to shift personnel including the authority and responsibility of the shift manager, licensed senior reactor operator in the control room, control room operator and shift technical advisor.

- Manipulation of controls and assignment of shift personnel to duty stations per the requirements of 10 CFR 50.54 (i), (j), (k), (l), and (m) including delineation of the space designated for the "At the Controls" area of the control room.
- Shift relief and turnover procedures.
- Fitness for Duty.
- Control Room access.
- Working hour limitations.
- Feedback of design, construction, and applicable important industry and operating experience.
- Shift Manager administrative duties.
- Verification of correct performance of operational activities.
- A vendor interface program that provides vendor information for safety related components is incorporated into plant documentation.
- Fire protection program implementation.
- A process for implementing the safety/security interface requirements of 10 CFR 73.58.

13.5.2 OPERATING AND MAINTENANCE PROCEDURES

13.5.2.1 Operating and Emergency Operating Procedures

This information is addressed in the DCD.

13.5.2.2 Maintenance and Other Operating Procedures

The QAPD, as described in Section 17.5, provides guidance for procedural adherence. Regulatory and industry guidance for the appropriate format, content, and typical activities delineated in written procedures is implemented as appropriate.

13.5.2.2.1 Plant Radiation Protection Procedures

The plant radiation protection program is contained in procedures. Procedures are developed and implemented for such things as: maintaining personnel exposures, plant contamination levels, and plant effluents ALARA; monitoring both external and internal exposures of workers, considering industry-accepted techniques; routine radiation surveys; environmental monitoring in the vicinity of the plant;

radiation monitoring of maintenance and special work activities; evaluation of radiation protection implications of proposed modifications; establishing quality assurance requirements applicable to the radiation protection program; and maintaining radiation exposure records of workers and others.

13.5.2.2.2 Emergency Preparedness Procedures

A discussion of emergency preparedness procedures can be found in the ESPA Emergency Plan.

13.5.2.2.3 Instrument Calibration and Test Procedures

The QAPD, as discussed in Section 17.5, provides a description of procedural requirements for instrumentation calibration and testing.

13.5.2.2.4 Chemistry Procedures

Procedures provided for chemical and radiochemical control activities include the nature and frequency of sampling and analyses; instructions for maintaining fluid quality within prescribed limits; the use of control and diagnostic parameters; and limitations on concentrations of agents that could cause corrosive attack, foul heat transfer surfaces or become sources of radiation hazards due to activation.

Procedures are also provided for the control, treatment, and management of radioactive wastes and control of radioactive calibration sources.

13.5.2.2.5 Radioactive Waste Management Procedures

Procedures for the operation of the radwaste processing systems provide for the control, treatment, and management of on-site radioactive wastes. Procedural controls are in place for radiological releases.

13.5.2.2.6 Maintenance, Inspection, Surveillance, and Modification Procedures

13.5.2.2.6.1 Maintenance Procedures

Maintenance procedures describe maintenance planning and preparation activities. Maintenance procedures are developed considering the potential impact on the safety of the plant, license limits, availability of equipment required to be operable, and possible safety consequences of concurrent or sequential maintenance, testing or operating activities.

Maintenance procedures contain sufficient detail to permit the maintenance work to be performed correctly and safely. Procedures include provisions for conducting and recording results of required tests and inspections, if not performed and documented under separate test and inspection procedures.

References are made to vendor manuals, plant procedures, drawings, and other sources as applicable.

Instructions are included, or referenced, for returning the equipment to its normal operating status. Testing is commensurate with the maintenance that has been performed. Testing may be included in the maintenance procedure or be covered in a separate procedure.

The preventive maintenance program, including preventive and predictive procedures, as appropriate for structures, systems and components, prescribes the frequency and type of maintenance to be performed. An initial program based on service conditions, experience with comparable equipment and vendor recommendations is developed prior to fuel loading. The program is revised and updated as experience is gained with the equipment. To facilitate this, equipment history files are created and kept current. The files are organized to provide complete and easily retrievable equipment history.

13.5.2.2.6.2 Inspection Procedures

The QAPD, as discussed in Section 17.5, provides a description of procedural requirements for inspections.

13.5.2.2.6.3 Modification Procedures

Plant modifications and changes to setpoints are developed in accordance with approved procedures. These procedures control necessary activities associated with the modifications such that they are carried out in a planned, controlled, and orderly manner. For each modification, design documents such as drawings, equipment and material specifications, and appropriate design analyses are developed or the as-built design documents are utilized. Separate reviews are conducted by individuals knowledgeable in both technical and QA requirements to verify the adequacy of the design effort.

Proposed modification(s) which involve a license amendment or a change to Technical Specifications are processed as proposed license amendment request(s).

Plant procedures impacted by modifications are changed prior to declaring the system operable to reflect revised plant conditions; and cognizant personnel who are responsible for operating and maintaining the modified equipment are adequately trained.

13.5.2.2.7 Material Control Procedures

The QAPD, as discussed in Section 17.5, provides a description of procedural requirements for material control.

13.5.2.2.8 Security Procedures

A discussion of security procedures is provided in the Security Plan.

The Special Nuclear Material (SNM) Physical Protection Program describes the 10 CFR Part 70 required protection program in effect for the period of time during which new fuel as SNM is received and stored in a controlled access area (CAA), in accordance with the requirements of 10 CFR 73.67.

The New Fuel Shipping Plan 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.

13.5.2.2.9 Special Nuclear Material (SNM) Material Control and Accounting Procedures

A material control and accounting system consisting of special nuclear material accounting procedures is utilized to delineate the requirements, responsibilities, and methods of special nuclear material control from the time special nuclear material is received until it is shipped from the plant. These procedures provide detailed steps for SNM shipping and receiving, inventory, accounting, and preparing records and reports. The Special Nuclear Material (SNM) Material Control and Accounting (MC&A) Program description is submitted to the Nuclear Regulatory Commission as a separate licensing basis document.

13.5.3 COMBINED LICENSE INFORMATION ITEM

STD COL 13.5-1
VEGP DEP 1.1-1

Information for this COL item is addressed in Section 13.5.

13.5.4 REFERENCES

201. ANSI/ANS 3.2-1988, “Administrative Control and Quality Assurance for the Operational Phase of Nuclear Power Plants.”

202. ANSI B30.2 (Chapter 2-3), “Overhead and Gantry Cranes.”

STD COL 13.5-1

Table 13.5-201
Pre-COL Phase Administrative Programs and Procedures

(This table is included for future designation as historical information.)

- Design/Construction Quality Assurance Program
- Reporting of Defects and Noncompliance, 10 CFR Part 21 Program
- Design Reliability Assurance Program

13.6 SECURITY

This section of the referenced DCD is incorporated by reference with the following departures and /or supplements.

This section of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

Add the following text after ESPA SSAR Section 13.6.

STD COL 13.6-1
STD COL 13.6-5

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) and is incorporated by reference (see Table 1.6-201). 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.

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 and is incorporated by reference (see Table 1.6-201). 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.

Table 13.4-201 provides milestones for security program and cyber security program implementation.

13.6.1 COMBINED LICENSE INFORMATION ITEM

STD COL 13.6-1

Information for the Security Plan portion of this COL item is addressed in Section 13.6.

Information for the Physical Security ITAAC portion of this COL item is addressed in Section 14.3.2.3.2.

STD COL 13.6-5

Information for the cyber security program portion of this COL item is addressed in Section 13.6.

13.6.2 ESP COL ACTION ITEMS

VEGP ESP COL 13.6-1 The specific access control measures to address the existing rail spur are addressed in Part 8, Physical Security Plan, Section 11.3.

13.6.3 REFERENCES

201. Not used.

VEGP DEP 1.1-1 DCD Section 13.7 is redistributed to include DCD Section 13.7 references 7, 8, and 10 with COLA FSAR Subsection 13.5.4 and DCD Section 13.7 references 2, 3, and 4 with COLA FSAR Subsection 13.6.3.

Add the following new section after DCD Section 13.6.

13.7 FITNESS FOR DUTY

This section of the referenced ESPA SSAR is incorporated by reference with the following variances and/or supplements.

Add the following text after ESPA SSAR Section 13.7 text.

STD SUP 13.7-1 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 (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.

VEGP SUP 13.7-1 The following site-specific information is provided:

- The construction site is defined in the Physical Security Plan, Appendix E and is under the control of Shaw Stone & Webster (Shaw). The 10 CFR Part 26 requirements are implemented for the construction site area based on the descriptions provided in Table 13.4-201.

- Construction Workers & First Line Supervisors (Shaw employees and subcontractors) are covered by the SNC-approved Shaw FFD Program (elements Subpart K).
- SNC employees and SNC subcontractor's construction management and oversight personnel are covered by the VEGP Units 1 and 2 Operations FFD Program and Shaw's employees and Shaw's subcontractors construction management and oversight personnel are covered by the SNC-approved Shaw FFD Program (elements Subpart A – H, N and O).
- SNC security personnel are covered by the VEGP Units 1 and 2 Operations FFD Program and Shaw's security personnel are covered by the SNC-approved Shaw FFD Program (elements Subpart A – H, N and O). This coverage is applicable from the start of construction activities to the earlier of (1) the receipt of SNM in the form of fuel assemblies, (2) the establishment of a protected area, or (3) the 10 CFR 52.103(g) finding.
- SNC FFD Program personnel are covered by the VEGP Units 1 and 2 Operations FFD Program and Shaw's FFD Program personnel are covered by the SNC approved Shaw FFD Program (elements Subpart A, B, D – H, N, O, and C per licensee's discretion).
- SNC security personnel protecting fuel assemblies, or the established protected area, or the facility following the 10 CFR 52.103(g) finding are covered by the VEGP Units 1 and 2 Operations FFD Program (elements Subpart A – I, N and O).
- Personnel required to physically report to the Technical Support Center (TSC) or Emergency Operations Facility (EOF) when that requirement is in effect are covered by the VEGP Units 1 and 2 Operations FFD Program (elements Subpart A – I, N, and O, except for §§ 26.205 – 209).

STD SUP 13.7-1 The operations phase FFD program is consistent with the applicable subparts of 10 CFR Part 26 (elements Subpart A – I, N, and O, except for individuals listed in §26.4(b), who are not subject to §§ 26.205 – 209.

13.7.1 REFERENCES

201. Nuclear Energy Institute "Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites," NEI 06-06, Revision 5, August 2009 (ML092430016).

Add the following new appendix at the end of DCD Chapter 13.

VEGP COL 13.1-1 APPENDIX 13AA CONSTRUCTION-RELATED ORGANIZATION

The information in this appendix is included for future designation as historical information. Paragraphs are numbered to be subsequent to Subsection 13.1.1.1.

13AA.1.1.1.1 Design and Construction Activities

The Westinghouse Electric Company (WEC) was selected to design, fabricate, deliver, and install the AP1000 advanced light water pressurized water reactors (PWR) and to provide technical direction for installation and startup of this equipment. DCD Subsection 1.4.1 provides detailed information regarding WEC past experience in design, development, and manufacturing of nuclear power facilities. Operating experience from design, construction, and operation of earlier WEC PWRs is applied in the design, construction, and operation of the AP1000 as described in numerous locations throughout the DCD (e.g., DCD Subsections 3.6.4.4, 3.9.4.2.1, 4.2.3.1.3).

Shaw provides the construction of the plant and additional design engineering for selected site specific portions of the plant. Shaw was selected based on experience and proven technical capability in nuclear construction projects or projects of similar scope and complexity.

Other design and construction activities are generally contracted to qualified suppliers of such services. Implementation or delegation of design and construction responsibilities is described in the subsections below. Quality assurance aspects of these activities are described in Chapter 17.

13AA.1.1.1.1.1 Principal Site-Related Engineering Work

The principal site engineering activities accomplished towards the construction and operation of the plant are:

a. Meteorology

Information concerning local (site) meteorological parameters is developed and applied by station and contract personnel to assess the impact of the station on local meteorological conditions. An onsite meteorological measurements program is employed by station personnel to produce data for the purpose of making atmospheric dispersion estimates for postulated accidental and expected routine airborne releases of effluents. A maintenance program is established for surveillance, calibration, and repair of instruments. More information regarding the study and meteorological program is found in Section 2.3.

b. Geology

Information relating to site and regional geotechnical conditions is developed and evaluated by utility and contract personnel to determine if geologic conditions could present a challenge to the safety of the plant. Items of interest include geologic structure, seismicity, geological history, and ground water conditions. During construction, foundations within the power block area are mapped or visually inspected and photographed. Section 2.5 provides details of these investigations.

c. Seismology

Information relating to seismological conditions is developed and evaluated by utility and contract personnel to determine if the site location and area surrounding the site is appropriate from a safety standpoint for the construction and operation of a nuclear power plant. Information regarding tectonics, seismicity, correlation of seismicity with tectonic structure, characterization of seismic sources, and ground motion are assessed to estimate the potential for strong earthquake ground motions or surface deformation at the site. Section 2.5 provides details of these investigations.

d. Hydrology

Information relating to hydrological conditions at the plant site and the surrounding area is developed and evaluated by utility and contract personnel. The study includes hydrologic characteristics of streams, lakes, shore regions, the regional and local groundwater environments, and existing or proposed water control structures that could influence flood control and plant safety. Section 2.4 includes more detailed information regarding this subject.

e. Demography

Information relating to local and surrounding area population distribution is developed and evaluated by utility and contract personnel. The data is used to determine if requirements are met for establishment of exclusion area, low population zone, and population center distance. Section 2.1 includes more detailed information regarding population around the plant site.

f. Environmental Effects

Monitoring programs are developed to enable the collection of data necessary to determine possible impact on the environment due to construction, startup, and operational activities and to establish a baseline from which to evaluate future environmental monitoring.

13AA.1.1.1.1.2 Design of Plant and Ancillary Systems

Responsibility for design and construction of systems outside the power block such as circulating water, service water, switchyard, and secondary fire protection systems are delegated to qualified contractors.

13AA.1.1.1.1.3 Review and Approval of Plant Design Features

Design engineering review and approval is performed in accordance with the reactor technology vendor QA program and Section 17.1. The reactor technology vendor is responsible for design control of the power block. Verification is performed by competent individuals or groups other than those who performed the original design. Design issues arising during construction are addressed and implemented with notification and communication of changes to the engineering director for review. As systems are tested and approved for turnover and operation, control of design is turned over to plant staff. The engineering director, along with functional managers and staff, assumes responsibility for review and approval of modifications, additions, or deletions in plant design features, as well as control of design documentation, in accordance with the Operational QA Program. Design control becomes the responsibility of the engineering director prior to loading fuel. During construction, startup, and operation, changes to human-system interfaces of control room design are approved using a human factors engineering evaluation addressed within Chapter 18. See Organization Charts, Figures 13.1-202 and 13AA-201 for reporting relationships.

13AA.1.1.1.1.4 Site Layout with Respect to Environmental Effects and Security Provisions

Site layout was considered when determining the expected environmental effects from construction.

The Physical Security Plan is designed with provisions that meet the applicable NRC regulations. Site layout was considered when developing the Security Plan.

13AA.1.1.1.1.5 Development of Safety Analysis Reports

Information regarding the development of the Final Safety Analysis Report is found in Chapter 1.

13AA.1.1.1.1.6 Review and Approval of Material and Component Specifications

Safety-related material and component specifications of structures, systems, and components designed by the reactor technology vendor are reviewed and approved in accordance with the reactor technology vendor quality assurance program and Section 17.1. Review and approval of items not designed by the reactor vendor are controlled for review and approval by Section 17.5 and the Quality Assurance Program Description.

13AA.1.1.1.1.7 Procurement of Materials and Equipment

Procurement of materials during the construction phase is the responsibility of the reactor technology vendor and constructor. The process is controlled by the construction QA programs of these organizations. Oversight of the inspection and receipt of materials process is the responsibility of the manager in charge of quality assurance.

13AA.1.1.1.1.8 Management and Review of Construction Activities

Overall management and responsibility for construction activities is assigned to the site vice president. The nuclear construction vice president is accountable to the site vice president for construction activities. See Organization Chart Figure 13AA-201. Monitoring and review of construction activities by utility personnel is a continuous process at the plant site. Contractor performance is monitored to provide objective data to utility management in order to identify problems early and develop solutions. Monitoring of construction activities verifies that contractors are in compliance with contractual obligations for quality, schedule, and cost. Monitoring and review of construction activities is divided functionally across the various disciplines of the utility construction staff, e.g. electrical, mechanical, instrument and control, etc., and tracked by schedule based on system and major plant components/areas.

After each system is turned over to plant staff the construction organization relinquishes responsibility for that system. At that time they will be responsible for completion of construction activities as directed by plant staff and available to provide support for preoperational and start-up testing as necessary.

Periodic assessment involving both the construction and operations organizations continues to identify SSCs that could reasonably be expected to be impacted by scheduled construction activities. Appropriate administrative and managerial controls are then established as necessary. Specific hazards, impacted SSCs, and managerial and administrative controls are reviewed on a recurring basis and, if necessary, controls are revised/developed and implemented and maintained current as work progresses on site. For example, prior to construction activities that involve the use of large construction equipment such as cranes, managerial and administrative controls are in place to prevent adverse impacts on any operating unit(s) overhead power lines, switchyard, security boundary, etc., by providing the necessary restrictions on the use of large construction equipment.

13AA.1.1.1.2 Preoperational Activities

The plant manager reports to the site vice president. The plant manager, with the aid of those managers that report directly to the plant manager, (see Figure 13AA-201) is responsible for the activities required to transition the unit from the construction phase to the operational phase. These activities include turnover of systems from construction, preoperational testing, schedule management, procedure development for tests, fuel load, integrated startup testing, and turnover of systems to plant staff.

During construction and initial testing, the Engineering, Procurement, and Construction (EPC) contractor is responsible for equipment maintenance. To ensure equipment operability and reliability, plant maintenance programs such as preventive and corrective maintenance are developed prior to system turnover and become effective as each system is turned over from the EPC contractor to the plant staff with approved administrative procedures under the direction of the managers in charge of maintenance, engineering, and work control.

13AA.1.1.1.2.1 Development of Human Factors Engineering Design Objectives and Design Phase Review of Proposed Control Room Layouts

Human factors engineering (HFE) design objectives are initially developed by the reactor technology vendor in accordance with Chapter 18 of the FSAR and the Design Control Document (DCD). As a collaborative team, personnel from the reactor technology vendor design staff and personnel, including licensed operators, engineers, and instrumentation and control technicians from owner and other organizations in the nuclear industry, assess the design of the control room and man-machine interfaces to attain safe and efficient operation of the plant. See Section 18.2 for additional details of HFE program management.

Modifications to the certified design of the control room or man-machine interface described in the Design Control Document are reviewed per engineering and site support procedures, as required by Section 18.2, to evaluate the impact to plant safety. The engineering director is responsible for the human factors engineering (HFE) design process and for the design commitment to HFE during construction and throughout the life of the plant as noted in Subsection 13.1.1.2.1. The HFE program is established in accordance with the description and commitments in Chapter 18.

13AA.1.1.1.2.2 Preoperational and StartupTesting

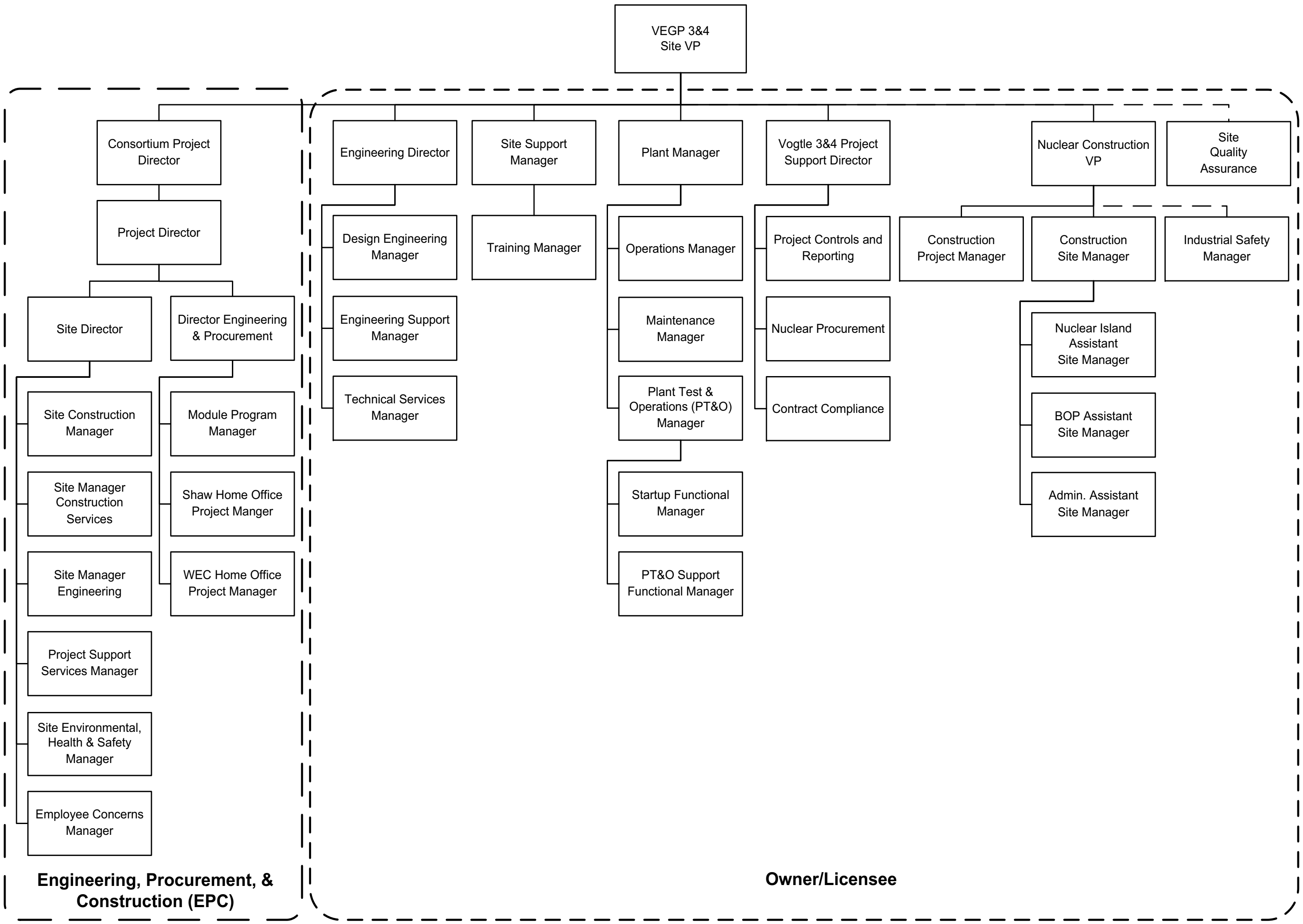
Preoperational and startup testing is conducted by the plant test and operations (PT&O) organization. The PT&O organization, functions, and responsibilities are addressed in Section 14.2. Sufficient numbers of personnel are assigned to perform preoperational and startup testing to facilitate safe and efficient implementation of the testing program. Plant-specific training provides instruction on the administrative controls of the test program. To improve operational experience, operations and technical staff are used as support in conducting the test program and in reviewing test results.

See Figure 13AA-201 for organization chart for preoperational and startup testing.

13AA.1.1.1.2.3 Development and Implementation of Staff Recruiting and Training Programs

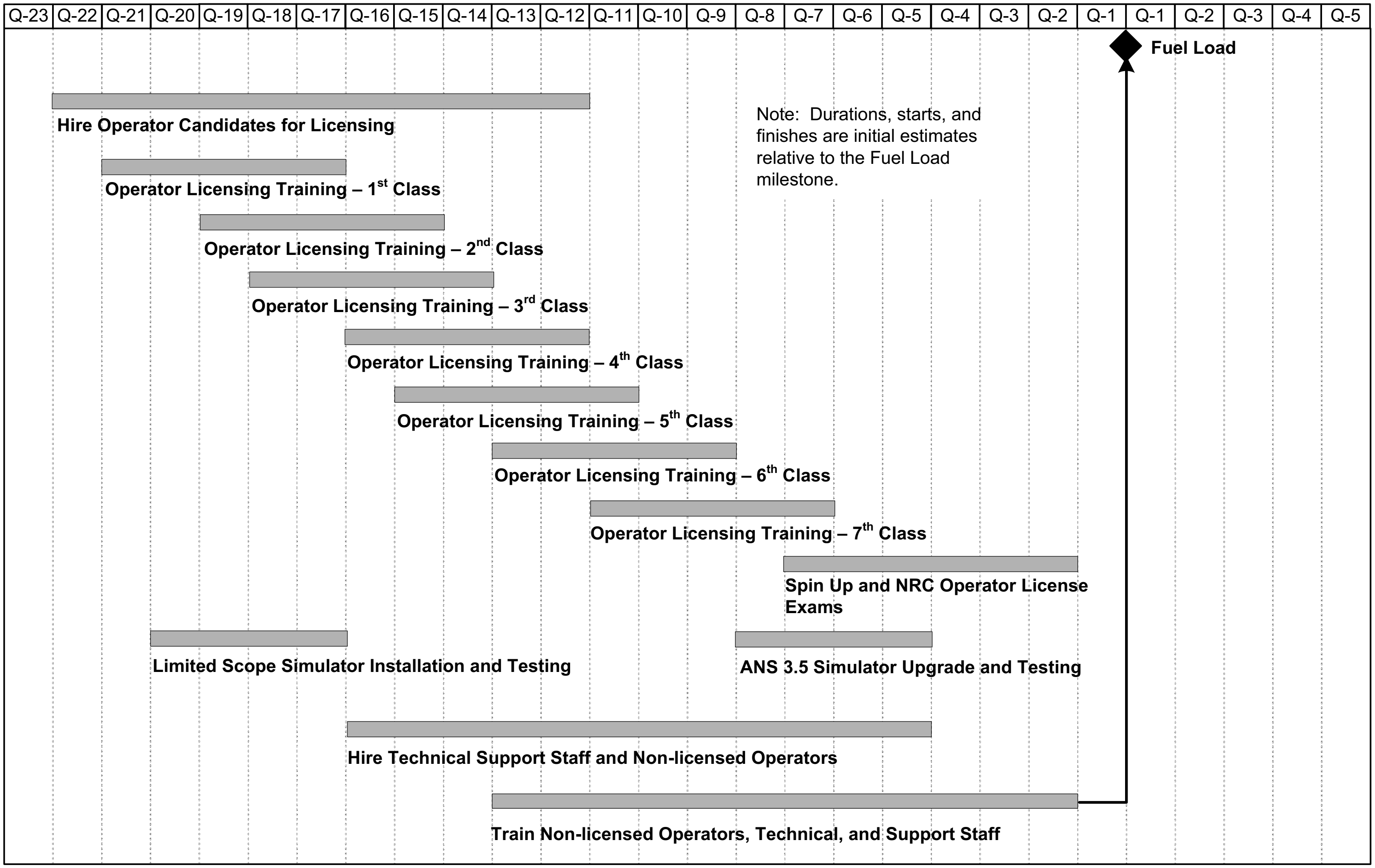
Staffing plans are developed based on operating plant experience with input from the reactor technology vendor for safe operation of the plant as determined by HFE. See Section 18.6. These plans are developed under the direction and guidance of the site vice president, vice president-engineering, and vice president-fleet operations support. Staffing plans are completed and manager level positions are filled prior to start of preoperational testing. Personnel selected to be licensed reactor operators and senior reactor operators along with other staff necessary to support the safe operation of the plant are hired with sufficient time available to complete appropriate training programs, and become qualified, and licensed, if required, prior to fuel being loaded in the reactor vessel. See Figure 13AA-202 for an estimated timeline of hiring requirements for operator and technical staff relative to fuel load.

Because of the dynamic nature of the staffing plans and changes that occur over time, it is expected that specific numbers of personnel on site will change; however, Table 13.1-201 includes the initial estimated number of staff for selected positions and the estimated number of additional positions required for a second unit. Recruiting of personnel to fill positions is the shared responsibility of the manager in charge of human resources and the various heads of departments. The training program is described in Section 13.2.



VEGP COL 13.1-1

Figure 13AA-201
Construction Management Organization



VEGP COL 13.1-1

Figure 13AA-202
Hiring Schedule for Plant Staff

CHAPTER 14

INITIAL TEST PROGRAM

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INITIAL TEST PROGRAM

14.1 SPECIFIC INFORMATION TO BE INCLUDED IN PRELIMINARY/FINAL SAFETY ANALYSIS REPORTS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

14.2 SPECIFIC INFORMATION TO BE INCLUDED IN STANDARD SAFETY ANALYSIS REPORTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

14.2.1 SUMMARY OF TEST PROGRAM AND OBJECTIVES

Add the following subsection at the end of DCD Subsection 14.2.1:

STD COL 14.4-3 FSAR Section 14.2 provides the requirements to be included in the Startup Administrative Manual (Procedures), as discussed in DCD Subsection 14.4.3. The information referenced in this section meets the Initial Test Program (ITP) criteria of NUREG-0800 and is formatted to follow Regulatory Guide 1.206, Part I, Section C.I.14.2.

The ITP is applied to structures, systems, and components that perform the functions described in the Regulatory Guide 1.68 evaluation in FSAR Section 1.9. The ITP is also applied to other structures, systems and components. The Startup Administrative Manual includes a list of the AP1000 structures, systems and components to which the ITP is applied.

Add the following Subsections after DCD Subsection 14.2.1.3

STD COL 14.4-3

14.2.1.4 Testing of First of a Kind Design Features

First of a kind (FOAK) testing may occur in any of the phases, depending on the nature of the testing and required sequencing of the tests. When testing FOAK design features, applicable operating experience from previous test performance on other AP1000 plants is reviewed, where available, and the ITP modified as needed based on those lessons learned.

14.2.1.5 Credit for Previously Performed Testing of First of a Kind Design Features

In some cases, FOAK testing is required only for the first of a new design or for the first few plants of a standard design. In such cases, credit may be taken for the previously performed tests. A discussion is included in the startup test reports of the results of those tests that are credited.

14.2.2 ORGANIZATION, STAFFING, AND RESPONSIBILITIES

Replace the existing information in DCD Subsection 14.2.2 with the following new paragraph and subsections.

STD COL 14.4-1 The AP1000 plant test and operations (PT&O) organization is described in Subsection 14.2.2.1. The organization for operating and maintaining the AP1000 plant is described in Section 13.1.

The PT&O organization structure (organizational chart) is included in the Startup Administrative Manual.

Table 13.4-201 provides milestones for initial test program implementation.

14.2.2.1 PT&O Organization

The Initial Test Program (ITP) is the responsibility of the PT&O Organization. The ITP includes three phases of testing:

- Construction and Installation Testing
- Preoperational Testing
- Startup Testing

14.2.2.1.1 Manager In Charge of PT&O

The manager in charge of PT&O reports directly to the plant manager. The manager in charge of PT&O manages the ITP. The manager in charge of PT&O is responsible for:

- Staffing the PT&O Organization.
- Developing, reviewing, and approving the administrative and technical procedures associated with the preoperational and startup phases.
- Managing the ITP and personnel.
- Implementing the ITP schedule.
- Managing contracts associated with the ITP.

14.2.2.1.2 Functional Manager In Charge of PT&O Support

The functional manager in charge of PT&O support reports directly to the manager in charge of PT&O. The functional manager in charge of PT&O support plans and schedules procedure development to support startup. The functional manager in charge of PT&O support verifies that the test documents conform to the approved project procedures.

The functional manager in charge of PT&O support reviews and approves test procedures. These procedures are used to demonstrate that a system and its components meet the design and performance criteria.

14.2.2.1.3 PT&O Engineers

The PT&O engineers report directly to the functional manager in charge of PT&O support. The PT&O engineers are responsible for developing system test procedures.

14.2.2.1.4 Functional Manager In Charge of Startup

The functional manager in charge of startup reports directly to the manager in charge of PT&O. The functional manager in charge of startup manages the preoperational and startup testing. The functional manager in charge of startup is responsible for:

- Participating in the Joint Test Working Group (JTWG) and ensuring that the JTWG reviews and approves administrative and test procedures. The JTWG structure and responsibilities are defined in Subsection 14.2.2.3.
- Preparing a detailed preoperational and startup testing schedule.
- Coordinating construction turnover to the PT&O organization.
- Informing the functional manager in charge of PT&O when vendor support essential to preoperational and startup testing is required, and coordinating vendor participation.
- Supervising and directing the startup engineers.
- Involving operations personnel in testing activities. Utilizing operations personnel, to the extent practical, as test witnesses or test performers to provide the operations personnel with experience and knowledge.
- Developing and implementing administrative controls to address system and equipment configuration control.
- Maintaining the startup schedule.
- Maintaining a daily startup log and issuing periodic progress reports that identify overall progress and potential challenges.

14.2.2.1.5 Startup Engineers

The startup engineers report directly to the functional manager in charge of startup. The startup engineers are responsible for:

- Complying with administrative controls.
- Identifying any special or temporary equipment or services needed to support testing.
- Coordinating testing with involved groups.
- Reviewing and evaluating test results.

14.2.2.2 PT&O Organization Personnel Qualifications and Training

Procedures are prepared to confirm that test personnel have adequate training, qualification and certification. Records are kept for extent of experience, involvement in procedure and test development, training programs, and level of qualification. The training organization qualifies Test Personnel as applicable, in accordance with the requirements of the applicable Quality Assurance Program. Training is performed as agreed between Westinghouse and the Licensee. Westinghouse test personnel training is per certified design.

Acceptable qualifications of non-supervisory test engineers follow the guidance provided in Regulatory Guide 1.28 as discussed in Appendix 1AA, i.e., ASME NQA-1-1994, Appendix 2A-1, Nonmandatory Guidance on the Qualification of Inspection and Test Personnel.

The training program/procedures shall include:

- The education, training, experience, and qualification requirements of supervisory personnel, test personnel, and other major participating organizations responsible for managing, developing, or conducting each test phase, or development of testing, operating, and emergency procedures.
- The establishment of a training program for each organizational unit, with regard to the scheduled preoperational and initial startup testing. This training program provides meaningful technical information beyond that obtained in the normal startup test program and provide supplemental operator training. This program also satisfies the criteria described in TMI Action Plan Item I.G.1 of NUREG-0660 and NUREG-0737.

The Startup Administrative Manual (Procedure) shall include:

- The implementation of measures to verify that personnel formulating and conducting test activities are not the same personnel who designed or are responsible for satisfactory performance of the system(s) or design features(s) being tested. This provision does not preclude members of the design organization from participating in test activities. This description also includes considerations of staffing effects that could result from overlapping initial test programs at multi-unit sites.

14.2.2.3 Joint Test Working Group

The Joint Test Working Group (JTWG) consists of an organizational group of authorized representative personnel from the Plant's operations and support group functions, Westinghouse Electric Company (WEC), Architect Engineer and other test support groups as identified below.

The Licensee has the overall responsibility for conduct of the Startup Test Program. The Westinghouse Startup Manager may be assigned overall responsibility and authority for technical direction of the Startup Test Program and may act as the JTWG Chairman.

The JTWG Chairman reports to the Chairman of the Plant Owner's Operations Review Committee (PORC) or qualified designee for matters of Startup test authority and acceptance.

The JTWG provides the following administrative oversight activities associated with the Startup Test Program:

- Review, evaluate and approve Startup Test Program administrative and test procedures.
- Oversee the implementation of the Preoperational Test Program and the Startup Test Program, including planning, scheduling and performance of Preoperational and Startup testing.
- Review and evaluate Construction, Preoperational and Startup test results and test turnover packages.

At a minimum, the JTWG is composed of qualified representatives provided from the following organizations:

- Licensee's Operations Group
- Licensee's Maintenance Group
- Site Preoperational Test Group
- Site Startup Test Group
- Licensee's Engineering Group

- Licensee's Corrective Action Organization
- Westinghouse Site Engineering Group
- Licensee's Health Physics/Chemistry Group
- Licensee's Quality Assurance Group

The following are additional generic details of the key responsibilities, authorities and interfaces of the Licensee Organizations delineated above:

- Operations Group

  The Operations Group has the overall responsibility for Plant Operations, including administrative control and tag-outs subsequent to system turnover. Their primary interfaces are with the Licensee Engineering and Technical Support organizations as well as the Westinghouse Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group.

- Maintenance Group

  The Maintenance Group has the overall responsibility for the Maintenance of Plant systems and components subsequent to System Turnover. They are key participants and maintainers of system maintenance control and tag-outs. Their primary interfaces are with the Licensee Operations Group and Technical Support organizations, as well as the Westinghouse Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group.

- Corrective Action Organization

  The Corrective Action Organization may be an organization specific to itself or may be a part of the Performance Assessment organization, the Quality Organization or another organization. This organization, together with every other site organization, is responsible for the administration and management of the corrective action program, as well as the identification of conditions adverse to quality. This organization interfaces with site organizations and identifies and documents conditions which need to be documented in the corrective action program.

- Engineering Group

  This group has the primary responsibility for site engineering and design oversight of the plant components and systems, as well as interfacing with the vendor engineering organization. This organization primarily interfaces with the Operations Group as well as the Westinghouse Site Engineering Organization, Preoperational and Startup Testing Teams and Construction

Services Group. The responsibility for training the testing personnel in accordance with applicable Quality Assurance Program is delegated and implemented as agreed to by Westinghouse. Westinghouse test personnel training is per certified design.

- Health Physics/Chemistry Group

  This Technical Support organization has the responsibility and authority to maintain Health Physics and system chemistry conditions at the plant, particularly after system turnover. This organization primarily interfaces with the Licensee Operations Group, as well as the Westinghouse Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group.

- Quality Assurance Group

  This group has the responsibility to verify that the applicable site Quality commitments are met within the scope of work performed at the site. This includes meeting the Criteria of 10 CFR 50 Appendix B. The primary interfaces for this group are the Licensee Operations Group and Technical Support organizations, including Quality Control and other quality organizations, as well as the Westinghouse Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group.

- Site Preoperational Test Group

  This group has the primary responsibility for the development, maintenance and performance of the site preoperational procedures at the site. The primary interfaces for this group are the Licensee Operations Group and Technical Support organizations, as well as the Westinghouse Engineering Organization, Startup Testing Teams and the Construction Services Group. Additional specific information regarding this organization’s responsibilities and interfaces is described in Subsection 14.2.2.5, below. Once preoperational testing is complete, this group turns systems over to the Startup Group.

- Site Startup Test Group

  This group has the primary responsibility for the development, maintenance and performance of the site startup procedures at the site. The primary interfaces for this group are the Licensee’s Operations Group and Technical Support organizations, as well as the Westinghouse Engineering Organization, Preoperational Testing Team and the Construction Services Group. Additional specific information regarding this organization’s responsibilities and interfaces is described in Subsection 14.2.2.6, below. The Startup Test Group turns over systems to the licensee when testing is complete.

- Westinghouse Site Engineering Group

  This group has the primary responsibility for the vendor interface between the site and the vendor's home offices, as well as the design authority for the primary vendor's components and systems. The various Westinghouse site leads for specific disciplines are a part of this organization. This organization primarily interfaces with Licensee Operations Group, as well as the Westinghouse Engineering Organization, Preoperational and Startup Testing Teams and Construction Services Group. The responsibility for training the testing personnel in accordance with the applicable Quality Assurance Program is delegated and implemented as agreed to by Westinghouse and the Licensee. Westinghouse test personnel training is per certified design.

14.2.2.4 Site Construction Group (Architect Engineer)

The Site Construction Group consists of the following, as necessary to support the Site Startup Test Program:

- Construction Group

  The Construction group has the primary responsibility for the construction and construction testing of the Balance of Plant (BOP) engineering systems and components. During Construction and Construction Testing, this group has authority over administrative control and tagouts of these systems. Their main interface is with the System Preoperational and Startup Testing Groups, as well as the Licensee Operations Group. The Construction Group is responsible for addressing open items in the system turnover punch lists to address turnover acceptability of the system.

- Construction Services Group

  The Construction Services Group primarily supports the Construction Group with activities necessary to support construction of systems and testing of the BOP systems and components, including the construction of scaffolding, installation and removal of insulation, and similar activities. With agreement between the necessary parties, this group may also support the Westinghouse Site Engineering Group with similar activities on the primary side. The primary interfaces of this group are the Construction Group and the organizations of the JTWG.

- Construction Services Procurement Group

  The Construction Services Procurement Group is responsible for the quality procurement of components and equipment necessary to support plant construction and testing. The primary interfaces of this group include the Construction Services Group and the Construction Services Quality Group.

- Construction Services Quality Group

  The Construction Services Quality Group is responsible for the oversight of the Quality Program during Construction Activities, including those pertinent to 10 CFR 50 Appendix B and the disposition of Significant Construction Deficiencies, 10 CFR 50.55(e) reports as necessary. This group primarily interfaces with the Construction and Services groups as well as the Westinghouse Site Engineering group and the JTWG.

- Construction Services Training Group

  This group is primarily responsible for the training and qualification of Site Construction Personnel in accordance with the applicable Quality Assurance Program. Their primary interface is with the qualified Construction personnel.

The Site Construction Group performs the following functions and scope of work, as necessary to support the Site Startup Test Program:

- Construction Installation and Testing, including management of construction testing documentation.
- Construction and Installation activities required to support Preoperational and Startup Test Programs.
- Vendor interface and procurement associated with supporting testing activities.
- Provide staffing as needed to support the testing activities.
- Turnover of Construction and Installation tested equipment, systems, and testing documentation to the Site Preoperational Test Group.

14.2.2.5 Site Preoperational Test Group

The Site Preoperational Test Group consists of the following, as necessary to support the Site Startup Test Program:

- Engineering Leads
- Preoperational Test Teams

The Site Preoperational Test Group performs the following functions and scope of work, as necessary to support the Site Startup Test Program:

- Coordinate tagging and maintenance prior to turnover to the Licensee to support system acceptance testing.

- Accept systems for turnover from the installation organization.
- Plan, scope and schedule plant systems for test to support the plant Preoperational Test Program.
- Manage and oversee the testing of plant systems to support the Plant Hot-Functional Test Program.
- Resolve open items and exceptions identified during implementation of the Preoperational Test Program.
- Accept and turn over Preoperational Test Packages to the Site Licensee.
- Support completion of Hot-Functional Test Program.
- Coordinate other support tasks required during Startup Testing activities with responsible groups (e.g., Licensee's Organization).

14.2.2.6 Site Startup Test Group

The Site Startup Test Group consists of the following, as necessary to support the Site Startup Test Program:

- Engineering Leads
- Startup Test Teams

The Site Startup Test Group performs the following functions and scope of work, as necessary to support the Site Startup Test Program:

- Coordinate tagging and maintenance as required to support system and equipment acceptance testing.
- Accept systems, structures and components from the Licensee for integrated testing.
- Plan, scope and schedule plant systems, structures and components for testing, to support Plant Startup.
- Manage and oversee the testing of plant systems, structures and components to support the Plant Power Ascension Test Program.
- Resolve open items and exceptions identified during implementation of the Startup Test Program.
- Accept and turn over Startup Test Packages to the Site Licensee.

- Coordinate other support tasks required during Startup Testing activities with responsible groups (e.g., Licensee's Organization).

14.2.3 TEST SPECIFICATIONS AND TEST PROCEDURES

Add the following text at the end of DCD Subsection 14.2.3:

STD COL 14.4-3 The Startup Administrative Manual shall include the following controls:

- Controls to provide test procedures that include appropriate prerequisites, objectives, safety precautions, initial test conditions, methods to direct and control test performance, and acceptance criteria by which the test is evaluated.

- Controls for the format of individual test procedures to provide consistency with the guidance contained in RG 1.68; or provide justifications for any exceptions.

- Controls to provide for participation of the principal design organizations in establishing test objectives, test acceptance criteria, and related performance requirements during the development of detailed test procedures. Each test procedure should include acceptance criteria that account for the uncertainties used in transient and accident analyses. The participating system designers should include the nuclear steam supply system vendor, architect-engineer, and other major contractors, subcontractors, and vendors, as applicable.

- Controls to provide for personnel with appropriate technical backgrounds and experience to develop and review test procedures. Persons filling designated management positions should perform final procedure review and approval.

- Controls to make the approved test procedures for satisfying FSAR testing commitments are made available to the NRC inspectors approximately 60 days prior to their intended use.

14.2.3.1 Conduct of Test Program

Add the following text and Subsection at the end of DCD Subsection 14.2.3.1:

STD COL 14.4-3 The Startup Administrative Manual (procedure) governs the initial testing and is issued no later than 60 days prior to the beginning of the pre-operational phase. Testing during all phases of the test program is conducted using approved test procedures.

14.2.3.1.1 Procedure Verification

Since procedures may be approved for implementation weeks or months in advance of the scheduled test date, a review of the approved test procedure is required before commencement of testing. The test engineer is responsible for verifying:

- Drawing and document revision numbers listed in the reference section of the test procedure agree with the latest revisions.
- The procedure text reflects any design and licensing (i.e., FSAR and Technical Specifications) changes made since the procedure was originally approved for implementation.
- Any new (since preparation of the procedure) Operating Experience lessons learned are incorporated into individual test procedures.

Procedures require signoff verification for prerequisites and instruction steps. This signoff includes identification of the person doing the signoff and the date and time of completion.

Test engineers maintain chronological logs of test status to facilitate turnover and aid in maintaining operational configuration control. These logs become part of the test documentation.

There is a documented turnover process to make known the test status and equipment configuration when personnel transfer responsibilities, such as during a shift change.

Test briefings are conducted for each test in accordance with administrative procedures. When a shift change occurs before test completion, another briefing occurs before resumption or continuation of the test.

Data collected is marked or identified with test, date, and person collecting data. This data becomes part of the test documentation.

The plant corrective action program is used to document deficiencies, discrepancies, exceptions, non-conformances and failures (collectively known as test exceptions) identified in the ITP. The corrective action documentation becomes part of the test documentation. WEC and/or other design organizations participate in the resolution of design-related problems that result in, or contribute to, a failure to meet test acceptance criteria.

The plant manager approves proceeding from one test phase to the next during the ITP. Approvals are documented in an overall ITP governance document.

Administrative procedures detail the test documentation review and approval. Review and approval of test documentation includes the test engineer, testing supervisor, Startup Group manager, WEC site representative or appropriate vendor, and JTWG. Final approval is by the plant manager.

Plant readiness reviews are conducted to assure that the plant staff and equipment are ready to proceed to the next test phase or plateau.

14.2.3.1.2 Work Control

STD SUP 14.2-5 The Startup Group is responsible for preparing work requests when assistance is required from the Construction organization. Work requests are issued in accordance with a site specific procedures governing the work management process. The plant staff, upon identifying a need for Construction organization assistance, coordinates their requirements through the appropriate Startup Test Engineer.

Activities requiring Construction organization work efforts are performed under the plant tagging procedures. Tagging requests are governed by a site-specific procedure for equipment clearance. Tagging procedures shall be used for protection of personnel and equipment and for jurisdictional or custodial conditions that have been turned over in accordance with the turnover procedure.

The Startup Group is responsible for supervising minor repairs and modifications, changing equipment settings, and disconnecting and reconnecting electrical terminations as stipulated in a specific test procedure. Startup Test Engineers may perform independent verification of changes made in accordance with approved test procedures.

14.2.3.1.3 System Turnover

STD SUP 14.2-6 During the construction phase, systems, subsystems, and equipment are completed and turned over in an orderly and well-coordinated manner. Guidelines are established to define the boundary and interface between related system/ subsystem and are used to generate boundary scope documents; for example, marked-up piping and instrument diagrams (P&IDs) and electrical schematic diagrams are provided for scheduling and subsequent development of component and system turnover packages. The system turnover process 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 of 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.

14.2.3.1.4 Conduct of Modifications During the Initial Test Program

STD SUP 14.2-7

Temporary alterations may be required to conduct certain tests. These alterations are documented in the test procedures. The test procedures contain restoration steps and retesting necessary to confirm satisfactory restoration to the required configuration. Modifications may be performed by the Construction organization or the plant staff processes prior to NRC issuance of the 10 CFR 52.103(g) finding. If the modification invalidates a previously completed ITAAC, then that ITAAC is re-performed. Each modification is reviewed to determine the scope of post-modification testing that is to be performed. Testing is conducted and documented to maintain the validity of preoperational testing and ITAAC. Alterations made following NRC issuance of the 10 CFR 52.103(g) finding are in accordance with plant processes and meet license conditions. Modifications that require changes to ITAAC require NRC approval of the ITAAC change.

14.2.3.1.5 Conduct of Maintenance During the Initial Test Program

STD SUP 14.2-8

Corrective or preventive maintenance activities are reviewed to determine the scope of postmaintenance testing to be performed. Prior to NRC issuance of the 10 CFR 52.103(g) finding, post-maintenance testing is conducted and documented to maintain validity of associated preoperational testing and ITAAC remain valid. Maintenance performed following NRC issuance of the 10 CFR 52.103(g) finding is in accordance with plant staff processes and meets license conditions.

14.2.3.2 Review of Test Results

Add the following Subsections at the end of DCD Subsection 14.2.3.2:

STD COL 14.4-4

14.2.3.2.1 Review and Approval Responsibilities

Upon completion of a test, the startup engineer is responsible for:

- Reviewing the test data.
- Evaluating the test results.
- Verifying that the acceptance criteria are met.
- Verifying that the test results that do not meet acceptance criteria are entered into the corrective action program.
- Verifying that the results of retesting do not invalidate ITAAC acceptance criteria.

Test results are reviewed and approved by the JTWG. Review and approval of test results are kept current such that succeeding tests are not dependent on systems or components that have not been adequately tested. Test exceptions which do not meet acceptance criteria are identified to the affected and responsible design organizations and entered into the corrective action program. Implementation of corrective actions and retests are performed as required.

Prior to initial fuel load, the results of the preoperational test phase are comprehensively reviewed by the PT&O organization and the JTWG to verify the results indicate that the required plant structures, systems, and components are capable of supporting the initial fuel load and subsequent startup testing. The plant manager approves fuel loading.

Each area of startup testing is reviewed and evaluated by the PT&O organization and the JTWG. The test results at each power ascension testing power plateau are reviewed and evaluated by the PT&O organization and the JTWG and approved by the plant manager before proceeding to the next plateau. Startup test reports are prepared in accordance with the guidance in position C.9 of Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.”

The reactor vendor is responsible for reviewing and approving the results of the tests of supplied equipment. Architect Engineer representatives review and approve the results of the tests of supplied equipment. Other vendors' representatives review and approve the results of the tests of supplied equipment.

Final approval of individual test completion is by the plant manager after approval by the Joint Test Working Group (JTWG).

14.2.3.2.2 Technical Evaluation

Each completed test package is reviewed by technically qualified personnel to confirm satisfactory demonstration of plant, system or component performance and compliance with design and license criteria.

14.2.3.3 Test Records

Add the following subsection at the end of DCD Subsection 14.2.3.3:

14.2.3.3.1 Startup Test Reports

STD COL 14.4-4 Startup test reports are generated describing and summarizing the completion of tests performed during the ITP. A startup report is submitted at the earliest of:

1) 9 months following initial criticality,

2) 90 days after completion of the ITP, or

3) 90 days after start of commercial operations. If one report does not cover all three events, then supplemental reports are submitted every three months until all three events are completed. These reports:

- Address each ITP test described in the FSAR.
- Provide a general description of measured values of operating conditions or characteristics obtained from the ITP as compared to design or specification values.
- Describe any corrective actions that were required to achieve satisfactory operation.
- Include any other information required by license conditions.

Add the following subsections after DCD Subsection 14.2.5:

Utilization of Operating Experience

STD SUP 14.2-4 Administrative procedures provide methodologies for evaluating and initiating action for operating experience information (OE). DCD Subsection 14.2.5 describes the general use of operating experience by WEC in the development of the test program.

14.2.5.1 Use of OE During Test Procedure Preparation

Administrative procedures require review of recent internal and external operating experience when preparing test procedures.

14.2.5.2 Sources and Types of Information Reviewed for ITP Development

Multiple sources of operating experience were reviewed to develop this description of the ITP administration program. These included INPO Reports, INPO Guidelines, INPO Significant Event Reports, INPO Significant Operating Experience Reports and NRC Regulatory Guide 1.68.

14.2.5.3 Conclusions from Review

The following conclusions are a result of the OE review conducted to develop this ITP administration program description:

- The test procedures should provide guidance as to the expected plant response and instructions concerning what conditions warrant aborting the test. Errors and problems with the procedures should be anticipated. A means for prompt but controlled approval of changes to test procedures is needed. Critical test procedures should provide specific criteria for test termination and specific steps to conduct termination is conducted in a safe and orderly manner. Providing procedural guidance for aborting the test could prevent delays in plant restoration. Conservative guidance for actions to be taken should be included in the procedures.

- Plant simulators may prove useful in preparing for special tests and verifying procedures.

- Appropriate component/system operability should be verified prior to critical tests.

- The need to perform physics tests that can produce severe power tilts should be evaluated, particularly if tests at other similar reactors have provided sufficient data to verify the adequacy of the nuclear physics analysis.

- Compensatory measures should be implemented in accordance with guidance for infrequently performed tests or evolutions, where appropriate.

14.2.5.4 Summary of Test Program Features Influenced by the Review

The conclusions from the preceding section were incorporated in Section 14.2.

14.2.5.5 Use of OE during Conduct of ITP

Administrative procedures require discussion of operating experience when performing pre-job briefs immediately prior to the conduct of a test.

14.2.6 USE OF PLANT OPERATING AND EMERGENCY PROCEDURES

Add the following text and Subsection to the end of DCD Subsection 14.2.6:

STD COL 14.4-3 These procedures are used extensively in the Human-Machine Interface Testing, which is integrated as a part of the Control Room Design finalization. Additionally, the AP1000 plant operating and emergency procedures are developed to support the following design finalization activities:

- Human Factors Engineering
- Operational Task Analysis
- Training Simulator Development
- Verification and Validation of the Procedures and the Training Material

The AP1000 emergency, abnormal and some normal operating procedures, along with some Alarm Response Procedures and surveillance procedures, are exercised and verified in the processes delineated above and in the Control Room design finalization process.

In addition, the AP1000 Preoperational Testing and Startup Test procedures are verified and validated during the design finalization process, which helps prevent human factors issues with the development of these procedures. In addition, the plant operators use the Normal Operating Procedures while preoperational and startup tests are performed, which adds to their validity and the plant operators training.

14.2.6.1 Operator Training and Participation during Certain Initial Tests (TMI Action Plan Item I.G.1, NUREG-0737)

The objective of operator participation is to increase the capability of shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and offnormal events is conducted.

Operators are trained on the specifics of the ITP schedule, administrative requirements and tests. Specific Just In Time training is conducted for selected startup tests.

The ITP may result in the discovery of an acceptable plant or system response that differs from the expected response. Test results are reviewed to identify these differences and the training for operators is changed to reflect them. Training is conducted as soon as is practicable in accordance with training procedures.

14.2.8 TEST PROGRAM SCHEDULE

Add the following text and subsection at the end of DCD Subsection 14.2.8:

STD SUP 14.2-1 A site-specific initial test program schedule will be provided to the NRC after issuance of the COL. This schedule will address each major phase of the test program (including tests that are required to be completed before fuel load), as well as the organizational impact of any overlap of first unit initial testing with initial testing of the second unit.

The sequential schedule for individual startup tests should establish that testing is completed in accordance with plant technical specification requirements for structures, systems and components (SSC) operability before changing plant modes. Additionally, the schedule establishes that the safety of the plant is not dependent on the performance of untested SSCs. Guidance provided in Regulatory Guide 1.68 is used for development of the schedule.

The Startup Administrative Manual shall include the following controls:

- Test Procedure Development Schedule:
    - Controls to establish a schedule for the development of detailed testing, plant operating, and emergency procedures. These procedures, to the extent practical, are trial-tested and corrected during the initial test program prior to fuel loading in order to establish their adequacy.
    - Controls to confirm that approved test procedures are in a form suitable for review by NRC inspectors at least 60 days prior to their intended use, or at least 60 days prior to fuel loading for fuel loading and startup test procedures.
    - Controls to provide timely notification to the NRC of changes in approved test procedures previously available for NRC review.

- Initial Test Program Schedule:
  - Controls to establish a schedule to conduct the major phases of the initial test program, relative to the expected fuel loading date. This is covered in License Conditions in Part 10 of the COL Application.
  - Controls to allow at least 9 months for conducting preoperational testing.
  - Controls to allow at least 3 months for conducting startup testing, including fuel loading, low-power tests, and power-ascension tests.
  - Controls to overlap test program schedules (for multi-unit sites) such that they do not result in significant divisions of responsibilities or dilutions of the staff provided to implement the test program.
  - Controls to sequence the schedule for individual startup tests, insofar as is practicable, such that testing is completed prior to exceeding 25 percent power for the plant SSCs that are relied upon to prevent, limit, or mitigate the consequences of postulated accidents. The schedule should establish that, insofar as is practicable, testing is accomplished as early in the test program as is feasible and that the safety of the plant is not dependent on the performance of untested SSCs.

The milestone schedule for developing plant operating procedures is presented in Table 13.4-201. The operating and emergency procedures are available prior to start of licensed operator training and, therefore, are available for use during the ITP. Required or desired procedure changes may be identified during their use. Administrative procedures describe the process for revising plant operating procedures.

14.2.9 PREOPERATIONAL TEST DESCRIPTIONS

Add the following subsection at the beginning of DCD Subsection 14.2.9

STD SUP 14.2-2 During preoperational testing, it may be necessary to return system control to Construction organization to repair or modify the system or to correct new problems. Administrative procedures include direction for:

- Means of releasing control of systems and or components to construction.

- Methods used for documenting actual work performed and determining impact on testing.
- Identification of required testing to restore the system to operability/functionality/availability status, and to identify tests to be re-performed based on the impact of the work performed.
- Authorizing and tracking operability and unavailability determinations.
- Verifying retests stay in compliance with ITAAC.

14.2.9.2.22 Pressurizer Surge Line Testing (First Plant Only)

STD COL 3.9-5

Purpose

The purpose of the pressurizer surge line testing is: a) to obtain data to verify the proper operation of temperature sensors installed on the pressurizer surge line and pressurizer spray line, and b) to obtain Reactor Coolant System piping displacement measurements for baseline data, as described in DCD subsections 3.9.3, 14.2.5, and 14.2.9.1.7 item (d).

Prerequisites

The construction tests for the individual components associated with the Reactor Coolant System have been completed. The testing and calibration of the required test instrumentation has been completed. The temporary sensors and instrumentation lead wires required for monitoring thermal stratification, cycling, and striping have been installed. The calibration of the transducers and the operability of the data acquisition equipment have been verified. Prior to testing of the piping system, a pretest walk-down shall be performed to verify that the anticipated piping movement is not obstructed by objects not designed to restrain the motion of the system (including instrumentation and branch lines). The system walk-down shall also verify that supports are set in accordance with the design.

General Test Methods and Acceptance Criteria

The performance of the Reactor Coolant System is observed and recorded during a series of individual tests that characterize the various modes of system operation. This testing verifies that the temperature sensors operate as described in DCD subsection 3.9.3 and in appropriate design specifications.

a) Verify the proper operation of temperature sensors installed on the pressurizer surge line and pressurizer spray line.

b) Record sensor data at specified intervals throughout hot functional testing of the RCS system, including during the drawing and collapsing of the bubble in the pressurizer.

c) Retain the following plant parameters time history for the same data recording period:

- Hot leg temperature
- Reactor Coolant System pressure
- Reactor coolant pump status
- Pressurizer level
- Pressurizer temperature (liquid and steam)
- Pressurizer spray temperature
- Pressurizer spray and auxiliary spray flow
- Normal residual heat removal system flow rate
- Passive core cooling system – passive residual heat removal flow rate.

d) Monitor pressurizer surge line and pressurizer spray line for valve leakage.

e) Remove the transducers and associated hardware after the completion of testing.

f) Proper operation of the temperature sensors in the pressurizer surge and spray lines is verified.

14.2.9.4.15 Seismic Monitoring System Testing

Add the following text at the beginning of DCD Subsection 14.2.9.4.15:

STD COL 14.4-5 The seismic monitoring system testing described in this section of the DCD also applies to site-specific seismic sensors.

Add the following subsections after DCD Subsection 14.2.9.4.21:

14.2.9.4.22 Storm Drains

STD COL 14.4-5 **Purpose**

Storm drain system testing verifies that the drains prevent plant flooding by diverting storm water away from the plant, as described in Section 2.4.

Prerequisites

Construction of the storm drain system is completed, and the system is operational.

General Test Methods and Acceptance Criteria

The storm drain system is visually inspected to verify the flow path is unobstructed. The system is observed under simulated or actual precipitation events to verify that the runoff from roof drains and the plant site and adjacent areas does not result in unacceptable soil erosion adjacent to, or flooding of, Seismic Category I structures.

14.2.9.4.23 Off-site AC Power Systems

Purpose

Off-site alternating current (ac) power system testing demonstrates the energization and proper operation of the as-installed switchyard components, as described in Section 8.2.

Prerequisites

Construction testing of plant off-site ac power systems, supporting systems, and components is completed. The components are operational and the switchyard equipment is ready to be energized. The required test instrumentation is properly calibrated and operational. The off-site grid connection is complete and available.

General Test Methods and Acceptance Criteria

The plant off-site ac power system components undergo a series of individual component and integrated system tests to verify that the off-site 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 off-site 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.

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 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.

The test results confirm that the off-site ac power systems meet the technical and operational requirements described in Section 8.2.

14.2.9.4.24 Raw Water System

Purpose

Raw water system testing verifies that the as-installed components supply raw water to the circulating water cooling tower basin, service water system cooling tower basin, fire protection water storage tanks, and other systems, as described in Subsection 9.2.11.

Prerequisites

Construction testing of the raw water system is completed. The components are operational and the storage tanks and cooling tower basins are able to accept water. Required support systems, electrical power supplies, and control circuits are operational.

General Test Methods and Acceptance Criteria

The raw water system component and integrated system performance is observed to verify that the system functions, as described in Subsection 9.2.11 and in appropriate design specifications. The individual component and integrated system tests include:

a. Operation of the system pumps, traveling screens, automatic strainers, and valves is verified.

b. Operation of the system instrumentation, controls, actuation signals, alarms, and interlocks is verified.

c. Operation of heat tracing on system piping is verified.

14.2.9.4.25 Sanitary Drainage System

Purpose

Sanitary drainage system testing verifies that the as-installed components properly collect and discharge sanitary waste, as described in DCD Subsection 9.2.6.

Prerequisites

Construction testing of the sanitary drainage system is completed. Required support systems, electrical power supplies, and control circuits are operational.

General Test Methods and Acceptance Criteria

The sanitary drainage system component and integrated system performance is observed to verify that the system functions, as described in Subsection 9.2.6.2.1 and in appropriate design specifications. The individual component and integrated system tests include:

a. Operation of lift stations and valves is verified.

b. Operation of the system instrumentation, controls, actuation signals, and interlocks is verified.

14.2.9.4.26 Fire Brigade Support Equipment

Purpose

Fire brigade support equipment testing verifies that the equipment operates and is available when needed to perform the fire brigade functions, as described in Section 9.5.

Prerequisites

Equipment is ready and available for testing.

General Test Methods and Acceptance Criteria

The fire brigade support equipment undergoes a series of inspections to verify availability and operability. Equipment is available for selection and use, based on the hazard. Fire brigade support equipment tests include:

a. Location of portable extinguishers is verified; portable extinguishers are verified fully charged.

b. Operation of portable ventilation equipment is verified.

c. Operation of portable communication equipment is verified.

d. Operation of portable lighting is verified.

e. Operation of self-contained breathing apparatus and face masks is verified.

f. Operation of keys to open locked fire area doors is verified.

g. Turnout gear functionality and availability is verified.

h. Compatibility of threads for hydrants, hose couplings, and standpipe risers with the local fire department equipment is verified, or alternatively, an adequate supply of readily available hose thread adaptors is verified.

14.2.9.4.27 Portable Personnel Monitors and Radiation Survey Instruments

Purpose

Portable personnel monitors and radiation survey instruments testing verifies that the devices operate in accordance with their intended function in support of the radiation protection program, as described in Chapter 12.

Prerequisites

Portable personnel monitors, radiation survey instruments, and appropriate certified test sources are on site.

General Test Method and Acceptance Criteria

The portable personnel monitors and radiation survey instruments are source checked, tested, maintained, and calibrated in accordance with the manufacturers' recommendations. The portable monitors and instruments tests include:

a. Proper function of the monitors and instruments to respond to radiation is verified, as required.

b. Proper operation of instrumentation controls, battery, and alarms, if applicable.

14.2.10 STARTUP TEST PROCEDURES

Add the following at the beginning of DCD Subsection 14.2.10:

STD SUP 14.2-3 The startup testing program is based on increasing power in discrete steps. Major testing is performed at discrete power levels as described in DCD Subsection 14.2.7. The first tests during Power Ascension Testing that verify movements and expansion of equipment are in accordance with design, and are conducted at a power level as low as practical (approximately 5 percent).

The governing Power Ascension Test Plan requires the following operations to be performed at appropriate steps in the power-ascension test phase:

- Conduct any tests that are scheduled at the test condition or power plateau.
- Confirm core performance parameters (core power distribution) are within expectations.
- Determine reactor power by heat balance, calibrate nuclear instruments accordingly, and confirm the existence of adequate instrumentation overlap between the startup range and power range detectors.
- Reset high-flux trips just prior to ascending to the next level to a value no greater than 20 percent beyond the power of the next level unless Technical Specification limits are more restrictive.
- Perform general surveys of plant systems and equipment to confirm that they are operating within expected values.
- Check for unexpected radioactivity in process systems and effluents.
- Perform reactor coolant leak checks.

- Review the completed testing program at each plateau; perform preliminary evaluations, including extrapolation core performance parameters for the next power level; and obtain the required management approvals before ascending to the next power level or test condition.

Upon completion of a given test, a preliminary evaluation is performed that confirms acceptability for continued testing. Smaller transient changes are performed initially, gradually increasing to larger transient changes. Test results at lower powers are extrapolated to higher power levels to determine acceptability of performing the test at higher powers. This extrapolation is included in the analysis section of the lower power procedure.

Surveillance test procedures may be used to document portions of tests, and ITP tests or portions of tests may be used to satisfy Technical Specifications surveillance requirements in accordance with administrative procedures. At Startup Test Program completion, a plant capacity warranty test is performed to satisfy the contract warranty and to confirm safe and stable plant operation.

Add the following subsection after DCD Subsection 14.2.10.4.28:

14.2.10.4.29 Cooling Tower(s)

STD COL 14.4-5 **Objectives**

- Verify proper cooling tower(s) function. Provide thermal acceptance testing of the cooling tower's heat removal capabilities.

Prerequisites

- The cooling tower(s) is structurally complete and in good operating condition.
- Circulating water system testing is complete.
- Required support systems, electrical power supplies, and control circuits are operational.

Test Method

Thermal performance of the cooling towers is tested and verified using established industry test standards.

Performance Criteria

The cooling tower(s) perform as described in Subsection 10.4.5 and in appropriate design specifications.

14.3 CERTIFIED DESIGN MATERIAL

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following subsections after DCD Subsection 14.3.2.2.

14.3.2.3 Site-Specific ITAAC (SS-ITAAC)

STD SUP 14.3-1 A table of inspections, tests, analyses, and acceptance criteria (ITAAC) entries is provided for each site-specific system described in this FSAR that meets the selection criteria, and that is not included in the certified design. The intent of these ITAAC is to define activities that are undertaken to verify the as-built system conforms with the design features and characteristics defined in the system design description. ITAAC are provided in tables with the following three-column format:

Design Commitment	**Inspection, Tests, Analyses**	**Acceptance Criteria**

Each design commitment in the left-hand column of the ITAAC tables has associated inspections, tests, or analyses (ITA) requirements specified in the middle column. The acceptance criteria for the ITA are defined in the right-hand column.

SS-ITAAC do not address ancillary buildings and structures on the site, such as administrative buildings, parking lots, warehouses, training facilities, etc.

Selection Criteria

- In determining those structures, systems, or components for which ITAAC must be prepared, the following questions are considered for each structure, system, or component:
  - Are any features or functions classified as Class A, B, or C?
  - Are any defense-in-depth features or functions provided?
  - For nonsafety-related systems, are any features or functions credited for mitigation of design basis events?
  - For nonsafety-related systems, are there any features or functions that have been identified in DCD Section 16.3 as candidates for additional regulatory oversight?

If the answer to any of the above questions is yes, then ITAAC are prepared.

- The scope and content of the ITAAC correspond to the scope and content of the site-specific system design description.
- One inspection, test, or analysis may verify one or more provisions in the system design description. An ITAAC that specifies a system functional test or an inspection may verify a number of provisions in the system design description. There is not necessarily a one-to-one correspondence between the ITAAC and the system design descriptions.
- As required by 10 CFR 52.103, the inspections, tests, and analyses are completed (and the acceptance criteria satisfied) prior to initial fuel loading.
- The ITAAC verify the as-built configuration and performance characteristics of structures, systems, and components as identified in the system design descriptions.

Selection Methodology – Using the selection criteria, ITAAC table entries are developed for each selected system. This is achieved by evaluating the design features and performance characteristics defined in the system design descriptions and preparing an ITAAC table entry for each design description criterion that satisfies the selection criteria. A close correlation exists between the left-hand column of the ITAAC and the corresponding design description entries.

The ITAAC table is completed by selecting the method to be used for verification (either a test, an inspection, or an analysis) and the acceptance criteria for the as-built feature.

The approach used to perform the tests, inspections, or analyses is similar to that described in DCD Subsection 14.3.2.2.

14.3.2.3.1 Emergency Planning ITAAC (EP-ITAAC)

VEGP SUP 14.3-1 EP-ITAAC were developed in the Early Site Permit (ESP) Application to address implementation of elements of the Emergency Plan. Site-specific EP-ITAAC are based on the generic ITAAC provided in Table 13.3-1 of SECY-05-0197. These ITAAC have been tailored to the specific reactor design and emergency planning program requirements.

14.3.2.3.2 Physical Security ITAAC (PS-ITAAC)

STD COL 13.6-1 Generic PS-ITAAC have been developed in a coordinated effort between the NRC and the Nuclear Energy Institute (NEI). These generic ITAAC have been tailored to the AP1000 design and site-specific security requirements.

14.3.2.3.3 Other Site-Specific Systems

STD SUP 14.3-1 One additional site-specific system has been determined to meet the ITAAC selection criteria, and ITAAC have been included for the Transmission Switchyard and Offsite Power System (ZBS) as indicated in Table 14.3-201. Systems not meeting the selection criteria are subject to the normal functional testing to verify that newly designed and installed systems, structures, or components perform as designed.

VEGP SUP 14.3-2 A summary of the AP1000 structures, systems, or components considered for selection is given in Table 14.3-201. This table supplements DCD Table 14.3-1.

14.3.3 CDM SECTION 3.0, NON-SYSTEM BASED DESIGN DESCRIPTIONS AND ITAAC

Add the following new subsection after the first paragraph in DCD Subsection 14.3.3

14.3.3.1 Non-System Based Site Specific ITAAC

VEGP SUP 14.3-3 Site specific ITAAC (SS-ITAAC) for the Nuclear Island engineered backfill and waterproof membrane are provided in ESPA SSAR Subsections 2.5.4.5.5 and 3.8.5 respectively.

14.3.3.2 Pipe Rupture Hazard Analysis ITAAC

STD COL 3.6-1 A pipe rupture hazard analysis is part of the piping design. The analyses will document that structures, systems, and components (SSCs) which are required to be functional during and following a design basis event have adequate high-energy and moderate-energy pipe break mitigation features. The locations of postulated ruptures and essential targets will be established and required pipe whip restraint and jet shield designs will be included. The as-designed pipe rupture hazards analysis will be based on the as-designed piping analysis and will be in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. The evaluation will address environmental and flooding effects of cracks in high and moderate energy piping. The report of the pipe rupture hazard analysis shall conclude that, for each postulated piping failure, the systems, structures, and components that are required to be functional during and following a design basis event are protected.

The as-built reconciliation of the pipe rupture hazards evaluation whip restraint and jet shield design in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5 are covered in as-built ITAAC identified in DCD Tier 1 to demonstrate that the as-built pipe rupture hazards mitigation features reflect the design, as reconciled. The reconciliation report will be made available for NRC inspection or audit when it has been completed.

The as-designed pipe rupture hazard 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.

14.3.3.3 Piping Design ITAAC

STD COL 3.9-7

The piping design ITAAC consists of the piping analysis for safety-related ASME Code piping. The piping design is completed on a package-by-package basis for applicable systems. In order to support closure of the piping design ITAAC, information consisting of the as-designed piping analysis for piping lines chosen to demonstrate all aspects of the piping design will be made available for NRC review, inspection, and/or audit. This information will consist of a design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class I piping. The piping packages to be analyzed are identified in the DCD.

The ASME Code prescribes certain procedures and requirements that are to be followed for completing the piping design. The piping design ITAAC includes a verification of the ASME Code design report to ensure that the appropriate code design requirements for each system's safety class have been implemented.

A reconciliation of the applicable safety-related as-built piping systems is covered in as-built ITAAC identified in DCD Tier 1 to demonstrate that the as-built piping reflects the design, as reconciled. The reconciliation report will be made available for NRC inspection or audit when it has been completed.

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.

VEGP SUP 14.3-2

Table 14.3-201
ITAAC Screening Summary

Structure/ System Acronym	Structure/System Description	Selected for ITAAC
DRS	Storm Drain System	XX
MES	Meteorological and Environmental Monitoring System	XX
RWS	Raw Water System	XX
TVS	Closed Circuit TV System	XX
VPS	Pump House Building Ventilation System	NA
YFS	Yard Fire Water System	XX
ZBS	Transmission Switchyard and Offsite Power System	XX
ZRS	Offsite Retail Power System	XX

Legend: XX = Site-specific system selected for ITAAC – title only, no entry for COLA

XX = Selected for ITAAC

NA = System is not part of VEGP design

14.4 COMBINED LICENSE APPLICANT RESPONSIBILITIES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

14.4.1 ORGANIZATION AND STAFFING

STD COL 14.4-1 This COL Item is addressed in Section 14.2.

14.4.2 TEST SPECIFICATIONS AND PROCEDURES

STD COL 14.4-2 Preoperational and startup test specifications and procedures are provided to the NRC in accordance with the requirements of DCD Subsection 14.2.3. The controls for development of test specifications and procedures are also described in Subsection 14.2.3.

A cross reference list is provided between ITAACs and test procedures and/or sections of test procedures.

14.4.3 CONDUCT OF TEST PROGRAM

STD COL 14.4-3 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 described in FSAR Section 14.2 is provided.

14.4.4 REVIEW AND EVALUATION OF TEST RESULTS

STD COL 14.4-4 Review and evaluation of individual test results, as well as final review of overall test results and selected milestones or hold points are addressed in Subsection 14.2.3.2. Test exceptions or results that do not meet acceptance criteria are identified to the affected and responsible design organizations, and corrective actions and retests, as required, are performed.

14.4.5 INTERFACE REQUIREMENTS

STD COL 14.4-5 This COL Item is addressed in Subsections 14.2.9.4.15, 14.2.9.4.22 through 14.2.9.4.27, 14.2.10.4.29, and in the Physical Security Plan.

14.4.6 FIRST-PLANT-ONLY AND THREE-PLANT-ONLY TESTS

STD COL 14.4-6 First-plant-only and first-three-plant-only tests either are performed in accordance with DCD Section 14.2.5 or a justification is provided that the results of the first-plant-only and first-three-plant-only tests are applicable to a subsequent plant. If the tests are not performed, the justification is provided prior to preoperational testing.

APPENDIX 14A
DESIGN ACCEPTANCE CRITERIA/ITAAC CLOSURE PROCESS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 15

ACCIDENT ANALYSES

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CHAPTER 15
ACCIDENT ANALYSES

15.0 ACCIDENT ANALYSES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.0.3.2 Initial Conditions

Add the following paragraph at the end of DCD Subsection 15.0.3.2.

STD COL 15.0-1 The plant operating instrumentation selected for feedwater flow measurement is a Caldon [Cameron] LEFM CheckPlus System (Reference 201), which will be calibrated (in a certified laboratory using a piping configuration representative of the plant piping design) prior to installation and will be tested after installation in the plant in accordance with the LEFM CheckPlus commissioning procedure. This selected plant operating instrumentation has documented instrumentation uncertainties to calculate a power calorimetric uncertainty that confirms the 1% uncertainty assumed for the initial reactor power in the safety analysis bounds the calculated calorimetric power uncertainty values. The calculated calorimetric is done in accordance with a previously accepted Westinghouse methodology (Reference 202). Administrative controls implement maintenance and contingency activities related to the power calorimetric instrumentation.

15.0.15 COMBINED LICENSE INFORMATION

Add the following text to the end of DCD Subsection 15.0.15.1.

STD COL 15.0-1 This COL item is addressed in FSAR Subsection 15.0.3.2.

15.0.16 REFERENCES

Add the following text to the end of DCD Subsection 15.0.16.

201. Final Safety Evaluation for Cameron Measurement Systems Engineering Report ER-157P, Revision 8, "Caldon Ultrasonics Engineering Report ER-157P, 'Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or Checkplus™ System'," (TAC No. ME1321). August 16, 2010. ADAMS Accession No. ML102160694.

202. Final Safety Evaluation for 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. September 24, 2001, ADAMS Accession No. ML012490569.

15.1 INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.5 INCREASE IN REACTOR COOLANT INVENTORY

This section of the referenced DCD is incorporated by reference with no departures or supplements.

15.6 DECREASE IN REACTOR COOLANT INVENTORY

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.6.5.3.7.3 Atmospheric Dispersion Factors

Add the following paragraph at the end of DCD Subsection 15.6.5.3.7.3.

VEGP COL 2.3-4 Site-specific χ/Q values provided in Section 2.3 are bounded by the values given in DCD Tables 15A-5 and 15A-6.

15.7 RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15.7.6 COMBINED LICENSE INFORMATION

VEGP COL 15.7-1 This COL item is addressed in ESPA SSAR Subsection 2.4.13.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 15A
EVALUATION MODELS AND PARAMETERS FOR ANALYSIS OF RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

15A.3.3 Atmospheric Dispersion Factors

Replace the third paragraph in DCD Subsection 15A.3.3 with the following:

VEGP COL 2.3-4 Site-specific χ/Q values provided in Section 2.3 are bounded by the values given in DCD Tables 15A-5 and 15A-6.

APPENDIX 15B
REMOVAL OF AIRBORNE ACTIVITY FROM THE CONTAINMENT ATMOSPHERE FOLLOWING A LOCA

This section of the referenced DCD is incorporated by reference with no departures or supplements.

CHAPTER 16

TECHNICAL SPECIFICATIONS

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CHAPTER 16
TECHNICAL SPECIFICATIONS

16.1 TECHNICAL SPECIFICATIONS

Subsections 16.1.1 and 16.1.2 of the DCD are incorporated by reference with no departures or supplements. The generic technical specifications and bases in Chapter 16 of the DCD are not considered Tier 2 information; therefore they are not incorporated by reference within this FSAR. However, the generic technical specifications and bases provided with Chapter 16 of the DCD are incorporated directly into the plant-specific technical specifications and bases provided in Part 4 of the COL application.

16.1.1 INTRODUCTION TO TECHNICAL SPECIFICATIONS

Combined License Information

VEGP COL 16.1-1 This COL Item (i.e., information addressing each of the remaining brackets [ ] in the AP1000 generic technical specifications) is addressed in Part 4 of the COLA.

16.2 DESIGN RELIABILITY ASSURANCE PROGRAM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

16.3 INVESTMENT PROTECTION

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

16.3.1 INVESTMENT PROTECTION SHORT-TERM AVAILABILITY CONTROLS

Add the following paragraph after the bulleted items at the end of the second paragraph of DCD Subsection 16.3.1:

STD COL 16.3-1 Station procedures govern and control the operability of investment protection systems, structures, and components, in accordance with Table 16.3-2 of the DCD, and provide the operating staff with instruction for implementing required actions when operability requirements are not met. Procedure development is addressed in FSAR Section 13.5.

16.3.2 COMBINED LICENSE INFORMATION

STD COL 16.3-1 This COL Item is addressed in Subsection 16.3.1.

CHAPTER 17

QUALITY ASSURANCE

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CHAPTER 17
QUALITY ASSURANCE

17.1 QUALITY ASSURANCE DURING THE DESIGN AND CONSTRUCTION PHASES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Replace the information in DCD Section 17.1 with the following information.

VEGP COL 17.5-1 SNC is responsible for the establishment and execution of quality assurance program requirements during the design, construction, and operations phases of VEGP Units 3 and 4. SNC may delegate the work of establishing and executing the quality assurance program, or any parts thereof, but retains responsibility for the quality assurance program.

Southern Nuclear has contracted with several vendors to develop the VEGP Units 3 and 4 COL application, including site characterization activities. The process of collection, review, and analysis of specific data was performed principally either under the Bechtel Corporation QA program or the MacTec QA program. SNC oversight is provided through its review and approval of these QA plans, by conducting QA audits and surveillances, and by direct participation in and oversight of the COL application development activities. The latter includes provide site specific applicant input and review of COL application content, signing the COL application as applicant at submittal, and working directly with these and other contractors to respond to NRC requests for additional information.

The “Quality Assurance Program Description” (QAPD) discussed in Section 17.5 establishes the QA program under which this COL application was developed and further established the requirements for the remaining portion of the design and construction phases.

17.2 QUALITY ASSURANCE DURING THE OPERATIONS PHASE

This section of the referenced DCD is incorporated by reference with no departures or supplements.

17.3 QUALITY ASSURANCE DURING DESIGN, PROCUREMENT, FABRICATION, INSPECTION, AND/OR TESTING OF NUCLEAR POWER PLANT ITEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

17.4 DESIGN RELIABILITY ASSURANCE PROGRAM

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD SUP 17.4-1 The quality assurance requirements for non-safety related SSCs within the scope of D-RAP is in accordance with the Quality Assurance Program Description (QAPD), Part III.

VEGP DEP 1.1-1

17.5 QUALITY ASSURANCE PROGRAM DESCRIPTION - NEW LICENSE APPLICANTS

VEGP COL 17.5-1
STD COL 17.5-2
STD COL 17.5-4
STD COL 17.5-8

The Quality Assurance Program in place during the design, construction, and operations phases is described in the QAPD, which is maintained as a separate document. This QAPD is incorporated by reference (see Table 1.6-201). This QAPD is based on NEI 06-14A, "Quality Assurance Program Description" (Reference 201).

Conformance statements for QA-related Regulatory Guides (including Regulatory Guides 1.28, 1.30, 1.33, 1.38, 1.39, 1.94, and 1.116) are provided in Appendix 1AA. While many Regulatory Guide positions can be identified as applicable to the scope of work identified and addressed by the DCD and others can be identified as applicable to the scope of work identified and addressed by the COLA, some QA guidance related positions could be accomplished by either scope of work and thus be addressed in either the DCD or the COLA. These positions are primarily dependent on who performs the work. The DCD conformance statement indicates an exception to apply NQA-1. The COLA identifies an exception to apply NQA-1. Per DCD Section 17.3, WEC work performed up to March 15, 2007 applied a 1991 version of the standard. A 1994 version of the standard is applied for work performed after that date by WEC. If the work is performed under the applicant's COL program, the 1994 version of NQA-1 identified in the COLA QAPD is applied. Thus, DCD scope (identified in DCD Appendix 1A) and "remaining scope" differentiate the application of the guidance identified in these Regulatory Guides.

VEGP COL 17.5-1

The QAPD is the SNC Nuclear Development Quality Assurance Manual.

STD COL 17.5-4

Table 13.4-201 provides milestones for operational quality assurance program implementation.

VEGP DEP 1.1-1

17.6 MAINTENANCE RULE PROGRAM

STD SUP 17.6-1
STD COL 3.8-5

This section incorporates by reference NEI 07-02A, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52" (Reference 202), with the following supplemental information. See Table 1.6-201.

Table 13.4-201 provides milestones for maintenance rule program implementation.

The text of the template provided in NEI 07-02A is generically numbered as "17.X." When the template is incorporated by reference into this FSAR, section numbering is changed from "17.X" to "17.6."

STD SUP 17.6-1

Descriptions of the programs listed in Subsection 17.6.3 of NEI 07-02A are provided in the following FSAR chapters/sections:

The maintenance rule program (Section 17.6)

The quality assurance program (Section 17.5)

Inservice inspection program (Sections 5.2 and 6.6)

Inservice testing program (Section 3.9)

The technical specifications surveillance test program (Chapter 16)

STD SUP 17.6-2

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.

VEGP DEP 1.1-1

17.7 COMBINED LICENSE INFORMATION ITEMS

Section 17.5 of the referenced DCD is incorporated by reference with the following departures and/or supplements.

VEGP COL 17.5-1 This COL Item is addressed in Sections 17.1 and 17.5.

STD COL 17.5-2 This COL Item is addressed in Section 17.5.

STD COL 17.5-4 This COL Item is addressed in Section 17.5.

STD COL 17.5-8 This COL Item is addressed in Section 17.5.

VEGP DEP 1.1-1

17.8 REFERENCES

Section 17.6 of the referenced DCD is incorporated by reference with the following departures and/or supplements.

---

201. Nuclear Energy Institute, Technical Report NEI 06-14A, “Quality Assurance Program Description,” Revision 7, July 2009.

202. Nuclear Energy Institute, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52,” NEI 07-02A, Revision 0, March 2008.

CHAPTER 18

HUMAN FACTORS ENGINEERING

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CHAPTER 18
HUMAN FACTORS ENGINEERING

18.1 OVERVIEW

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.2 HUMAN FACTORS ENGINEERING PROGRAM MANAGEMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

18.2.1.3 Applicable Facilities

Add the following information at the end of DCD subsection 18.2.1.3

VEGP COL 18.2-2 The EOF and TSC communications strategies, as well as the EOF and TSC Human Factors attributes, are described in the Emergency Plan. Subsection 9.5.2.2.5 provides additional information related to offsite interfaces.

18.2.6 COMBINED LICENSE INFORMATION

18.2.6.2 Emergency Operations Facility

VEGP COL 18.2-2 This COL item is addressed in Section 18.2.1.3.

18.3 OPERATING EXPERIENCE REVIEW

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.4 FUNCTIONAL REQUIREMENTS ANALYSIS AND ALLOCATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.5 AP1000 TASK ANALYSIS IMPLEMENTATION PLAN

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.6 STAFFING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD COL 18.6-1 Replace the DCD paragraph in Section 18.6 with the following information.

Table 13.1-201 contains the estimated staffing levels for those categories of personnel that are addressed by the Human Factors Engineering program per NUREG-0711, “Human Factors Engineering Program Review Model” (Reference 201), as follows:

- Licensed operators
- Shift Supervisors
- Non-licensed operators
- Shift technical advisors
- Instrumentation and control technicians
- Mechanical maintenance technicians
- Electrical maintenance technicians
- Radiation protection technicians
- Chemistry technicians
- Engineering support

The minimum level of staffing for control room personnel who directly monitor and control the plant is stated in Table 13.1-202 and meets the requirements of 10 CFR 50.54(m). Information about the staffing levels of security personnel is contained in the separately submitted physical security plan.

Qualification requirements of plant personnel listed above are discussed in Subsections 13.1.1.4, Qualifications of Technical Support Personnel, and 13.1.3, Qualification Requirements of Nuclear Plant Personnel, and, for security personnel, in the physical security plan.

The baseline level of staffing for the categories of personnel discussed above is derived from experience in current operating nuclear power plants. The number of personnel in operating plants has evolved over many years to a level that is safe and efficient and provides adequate personnel to operate the plant under all conditions, including abnormal and emergency, meets regulatory requirements, and supports individual training and personal needs.

Iterative adjustments are implemented to the level of staffing, as necessary, based on findings and input from periodic reviews and staffing analysis. Input to this analysis includes information derived from the other elements of the human factors engineering program, particularly operating experience review, functional requirements analysis and function allocation, task analysis, human reliability analysis, human-system interface design, procedure development, and training program development.

In addition to the regulatory requirements referenced, input to the analyses and the level of staffing is provided by WCAP-14694, “Designer’s Input to Determination of the AP600 Main Control Room Staffing Level” (DCD Section 18.6, Reference 1), AP1000 Combined License Technical Report APP-GW-GLR-010, “AP1000 Main Control Room Staff Roles and Responsibilities” (Reference 202), and EPRI Technical Report 1011717, “Program on Technology Innovation: Staff Optimization Scoping Study for New Nuclear Power Plants” (Reference 203).

18.6.1 COMBINED LICENSE INFORMATION ITEM

STD COL 18.6-1 This COL Item is addressed in Section 18.6.

18.6.2 REFERENCES

201. United States Nuclear Regulatory Commission, “Human Factors Engineering Program Review Model,” NUREG-0711, Revision 2, February 2004.

202. Westinghouse, “AP1000 Main Control Room Staff Roles and Responsibilities,” APP-GW-GLR-010, Rev. 2, June 2007.

203. EPRI, “Program on Technology Innovation: Staff Optimization Scoping Study for New Nuclear Power Plants,” Technical Report 1011717, Final Report, August 2005.

18.7 INTEGRATION OF HUMAN RELIABILITY ANALYSIS WITH HUMAN FACTORS ENGINEERING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.8 HUMAN SYSTEM INTERFACE DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

18.8.3.5 Technical Support Center Mission and Major Tasks

Replace the last sentence of the first paragraph with the following:

VEGP DEP 18.8-1
VEGP ESP PC 8

The Technical Support Center (TSC) location is described in the Emergency Plan.

18.8.3.6 Operations Support Center Mission and Major Tasks

Replace the last sentence of the first paragraph with the following:

VEGP DEP 18.8-1

The Operations Support Center (OSC) location is described in the Emergency Plan.

18.9 PROCEDURE DEVELOPMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.10 TRAINING PROGRAM DEVELOPMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Add the following paragraphs at the end of DCD subsection 18.10:

STD COL 18.10-1 Information regarding training program development is located in Section 13.2, Training. The training organization and roles and responsibilities of training personnel are discussed in Section 13.1, Organizational Structure of Applicant.

18.10.1 COMBINED LICENSE INFORMATION

STD COL 18.10-1 This COL Item is addressed in Section 18.10, 13.1 and 13.2.

18.11 HUMAN FACTORS ENGINEERING VERIFICATION AND VALIDATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.12 INVENTORY

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.13 DESIGN IMPLEMENTATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

18.14 HUMAN PERFORMANCE MONITORING

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

Replace the DCD paragraph with the following text.

STD COL 18.14-1 Human performance monitoring applies after the plant is placed in operation. The human performance monitoring process implements the guidance and methods as described in DCD Section 18.14 Reference 1.

The human performance monitoring process provides reasonable assurance that:

- The design can be effectively used by personnel, including within the control room and between the control room and local control stations and support centers.
- Changes made to the human system interface(s), procedures, and training do not have adverse effects on personnel performance, (e.g., a change does not interfere with previously trained skills).
- Human actions can be accomplished within time and performance criteria.
- The acceptable level of performance established during the design integrated system validation is maintained.

The human performance monitoring process is structured such that:

- Human actions are monitored commensurate with their safety importance.
- Feedback of information and corrective actions are accomplished in a timely manner.
- Degradation in performance can be detected and corrected before plant safety is compromised (e.g., by use of the plant simulator during training exercises).

The human performance monitoring process for risk-informed changes is integrated into the corrective action program, training program and other programs as appropriate. Identified human performance conditions/issues are evaluated for human factors engineering applicability.

Human factors engineering conditions are assigned specific human factors cause determination codes, trended for indications of degraded performance or potential human performance failures and have specific corrective actions identified.

The cause investigation:

- Identifies the cause of the failure or degraded performance to the extent that corrective action can be taken consistent with the corrective action program requirements.
- Addresses failure significance which includes the circumstances surrounding the failure or degraded performance, the characteristics of the failure, and whether the failure is isolated or has generic or common cause implications.
- Identifies and establishes corrective actions necessary to preclude the recurrence of unacceptable failures or degraded performance in the case of a significant condition adverse to quality.

When appropriate, design changes are integrated into training exercises to monitor for degradation in performance and allow for early detection and corrective actions before plant safety is challenged (e.g., by use of the plant simulator during training exercises).

Plant or personnel performance under actual design conditions may not be readily measurable. When actual conditions cannot be simulated, monitored, or measured, the available information that most closely approximates performance data in actual conditions should be used.

Monitoring strategies for human performance trending after the implementation of design changes is capable of demonstrating that performance is consistent with that assumed in the various analyses conducted to justify the change.

Risk-informed changes are screened commensurate with their safety importance to determine if the change requires monitoring of actions. For changes which require monitoring, the appropriate monitoring requirements are determined and implemented in the training program or other program as appropriate.

CHAPTER 19

PROBABILISTIC RISK ASSESSMENT

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CHAPTER 19
PROBABILISTIC RISK ASSESSMENT

19.1 INTRODUCTION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.2 INTERNAL INITIATING EVENTS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.3 MODELING OF SPECIAL INITIATORS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.4 EVENT TREE MODELS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.5 SUPPORT SYSTEMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.6 SUCCESS CRITERIA ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.7 FAULT TREE GUIDELINES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.8 PASSIVE CORE COOLING SYSTEM - PASSIVE RESIDUAL HEAT REMOVAL

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.9 PASSIVE CORE COOLING SYSTEM - CORE MAKEUP TANKS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.10 PASSIVE CORE COOLING SYSTEM - ACCUMULATOR

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.11 PASSIVE CORE COOLING SYSTEM - AUTOMATIC DEPRESSURIZATION SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.12 PASSIVE CORE COOLING SYSTEM - IN-CONTAINMENT REFUELING WATER STORAGE TANK

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.13 PASSIVE CONTAINMENT COOLING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.14 MAIN AND STARTUP FEEDWATER SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.15 CHEMICAL AND VOLUME CONTROL SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.16 CONTAINMENT HYDROGEN CONTROL SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.17 NORMAL RESIDUAL HEAT REMOVAL SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.18 COMPONENT COOLING WATER SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.19 SERVICE WATER SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.20 CENTRAL CHILLED WATER SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.21 AC POWER SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.22 CLASS 1E DC & UPS SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.23 NON-CLASS 1E DC & UPS SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.24 CONTAINMENT ISOLATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.25 COMPRESSED AND INSTRUMENT AIR SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.26 PROTECTION AND SAFETY MONITORING SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.27 DIVERSE ACTUATION SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.28 PLANT CONTROL SYSTEM

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.29 COMMON CAUSE ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.30 HUMAN RELIABILITY ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.31 OTHER EVENT TREE NODE PROBABILITIES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.32 DATA ANALYSIS AND MASTER DATA BANK

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.33 FAULT TREE AND CORE DAMAGE QUANTIFICATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.34 SEVERE ACCIDENT PHENOMENA TREATMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.35 CONTAINMENT EVENT TREE ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.36 REACTOR COOLANT SYSTEM DEPRESSURIZATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.37 CONTAINMENT ISOLATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.38 REACTOR VESSEL REFLOODING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.39 IN-VESSEL RETENTION OF MOLTEN CORE DEBRIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.40 PASSIVE CONTAINMENT COOLING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.41 HYDROGEN MIXING AND COMBUSTION ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.42 CONDITIONAL CONTAINMENT FAILURE PROBABILITY DISTRIBUTION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.43 RELEASE FREQUENCY QUANTIFICATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.44 MAAP4.0 CODE DESCRIPTION AND AP1000 MODELING

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.45 FISSION PRODUCT SOURCE TERMS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.46 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.47 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.48 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.49 OFFSITE DOSE EVALUATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.50 IMPORTANCE AND SENSITIVITY ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.51 UNCERTAINTY ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.52 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.53 NOT USED

This section was not required for DCD and is not used by DCD and FSAR.

19.54 LOW POWER AND SHUTDOWN PRA ASSESSMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.55 SEISMIC MARGIN ANALYSIS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

19.55.6.3 Site Specific Seismic Margin Analysis

VEGP COL 19.59.10-6 The VEGP GMRS exceeds the AP1000 CSDRS over certain frequency ranges. Site-specific seismic soil-structure interaction analyses of the AP1000 Nuclear Island (NI) using the VEGP GMRS ground motion and VEGP soil profile demonstrated that the resulting VEGP in-structure response spectra (ISRS) at the six key locations are enveloped by the AP1000 CSDRS broadened ISRS by a significant margin except for minor exceedances at very low frequency ranges. These slight exceedances have been shown to have no impact on the NI structures, systems, and components. Therefore, the Tier 1 criteria for SSE have been satisfied. This evaluation is presented in Section 3.7. In regards to seismic demand for the NI structures, systems, and components, it can be concluded that the Seismic Margin Assessment analysis documented in Section 19.55 is applicable to the VEGP site.

For seismic stability of the NI with regards to sliding and overturning, it was demonstrated that VEGP NI margins against sliding and overturning were greater than the limiting margins calculated for the standard AP1000 design cases. For seismic stability, it can be concluded that the Seismic Margin Assessment analysis documented in DCD Section 19.55 is applicable to the VEGP site.

For site specific conditions relating to soil-related failure modes, the demonstration of adequate seismic margin of the AP1000 design at the VEGP site is performed for a review level earthquake of 1.67 x VEGP GMRS, where the VEGP site-specific review level earthquake seismic responses and seismic loads are defined as 1.67 x VEGP GMRS seismic responses and seismic loads.

Potential for soil liquefaction was evaluated at 1.67 x VEGP GMRS which produces a peak ground acceleration of 0.44g. The liquefaction potential factor of safety was found to be high such that liquefaction potential was screened out as a contributor to design-specific plant-level HCLPF capacity. Similarly, bearing pressure capacity to demand still demonstrated sufficient margin so this potential failure mode was screened out as a contributor to design specific plant-level HCLPF capacity.

19.56 PRA INTERNAL FLOODING ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.57 INTERNAL FIRE ANALYSIS

This section of the referenced DCD is incorporated by reference with no departures or supplements.

19.58 WINDS, FLOODS, AND OTHER EXTERNAL EVENTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

19.58.3 CONCLUSION

Add the following information at the end of DCD Subsection 19.58.3:

VEGP SUP 19.58-1 Table 19.58-201 documents the site-specific external events evaluation that has been performed for VEGP Units 3 and 4. This table provides a general explanation of the evaluation and resultant conclusions and provides a reference to applicable sections of the COL where more supporting information (including data used, methods and key assumptions) regarding the specific event is located. Based upon this evaluation, it is concluded that the VEGP Units 3 and 4 site is bounded by the High Winds, Floods and Other External Events analysis documented in DCD Section 19.58 and APP-GW-GLR-101 (Reference 201) and no further evaluations are required at the COL application stage.

19.58.4 REFERENCES

201. Westinghouse Electric Company LLC, “AP1000 Probabilistic Risk Assessment Site-Specific Considerations,” Document Number APP-GW-GLR-101, Revision 1, October 2007.

VEGP SUP 19.58-1

Table 19.58-201 (Sheet 1 of 10)
External Event Screening for VEGP

Category	Event	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency
High Winds	F0 Tornado	Y	The tornado strike probability for the VEGP site area is discussed in VEGP ESPA SSAR Subsection 2.3.1.3.2. Vogtle has conservatively assumed that the strike probability for a tornado of a given intensity is equal to the overall strike probability for any tornado. Since the event frequencies are all greater than 1E-07, this event is applicable to the VEGP site. These event frequencies are bounded by the limiting initiating event frequencies given in Table 3.0-1 of APP-GW-GLR-101. Also, as documented in FSAR Table 2.0-201 the VEGP site characteristic tornado wind loadings are equal to the AP1000 DCD site characteristic tornado wind loadings. Therefore, the safety features of the AP1000 are unaffected and the CDFs given in APP-GW-GLR-101 Table 3.0-1 for these events are applicable to VEGP Units 3 and 4.	7.74E-05
	F1 Tornado	Y		7.74E-05
	F2 Tornado	Y		7.74E-05
	F3 Tornado	Y		7.74E-05
	F4 Tornado	Y		7.74E-05
	F5 Tornado	Y		7.74E-05

VEGP SUP 19.58-1

Table 19.58-201 (Sheet 2 of 10)
External Event Screening for VEGP

Category	Event	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency
High Winds (cont.)	Cat. 1 Hurricane	Y	Tropical cyclones are discussed in VEGP ESPA SSAR Subsection 2.3.1.3.3. The event frequencies are based on the number of recorded events over the 154 year period of record. There were no recorded events for Category 4 or 5 hurricanes. However a conservative event frequency of <1E-02 was assigned for these events. These event frequencies were provided to Westinghouse during the development of APP-GW-GLR-101. In 3 of the categories (Cat. 1, 3 Hurricanes and Extra-tropical storms) the event frequencies slightly exceed those given in Table 3.0-1 of APP-GW-GLR-101. This has been attributed to rounding of the values originally provided to Westinghouse by SNC. This change does not impact the conclusion in APP-GW-GLR-101 that none of the limiting event frequencies are sufficiently low to be removed from further consideration. As documented in FSAR Table 2.0-201 the VEGP site characteristic tornado wind loadings are equal to the AP1000 DCD site characteristic tornado wind loadings. The VEGP site characteristic operating basis wind speed (104 mph) is below the DCD site characteristic operating basis wind speed of 145 mph. Therefore, it is concluded that the safety features of the AP1000 are unaffected and the resulting CDFs given in AP-GW-GLR-101 Table 3.0-1 for these events are bounding to VEGP Units 3 and 4. Winds below 74 mph (tropical storms, depressions) are not considered to have an adverse impact on VEGP Units 3 and 4 as the switchyard and non-safety buildings will be designed to function at a higher wind speed (104 mph as discussed above). Therefore, no additional PRA considerations are required for winds below hurricane force.	1.04E-01
	Cat. 2 Hurricane	Y		2.60E-02
	Cat. 3 Hurricane	Y		3.25E-02
	Cat. 4 Hurricane	Y		<1E-02
	Cat. 5 Hurricane	Y		<1E-02
	Extra-tropical storms	Y		3.25E-02

VEGP SUP 19.58-1

Table 19.58-201 (Sheet 3 of 10)
External Event Screening for VEGP

Category	Event	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency
External Flood	External flood	N	External flooding is addressed in Section 2.4 of the COLA FSAR and ESPA SSAR. The design basis flood event for flooding from the Savannah River is described in SSAR Section 2.4.2.2 and is based on dam failures coincident with wind set-up and wave run-up. The design basis flood level derived is El. 178.10 ft. msl which is well below the site grade of 220 ft msl. Flooding due to a local Probable Maximum Precipitation (PMP) event is addressed in FSAR Subsections 2.4.2 and 2.4.10. The maximum water level in the power block area due to this event is 219.47 ft msl, which is below the entrance and openings to all safety related structures (elevation 220 ft msl). The PMP is the maximum rainfall that can physically occur at the site and the analysis performed contains significant conservatisms such that this value represents a bounding maximum flood elevation. As discussed in COLA FSAR Subsection 1.2.2, the VEGP site grade elevation of 220 ft. msl corresponds to DCD grade elevation 100 ft. Because no external flooding event exceeds this elevation it is concluded that the VEGP site is not susceptible to any external floods which would adversely impact safe operation of VEGP Units 3 and 4. The site is within the bounds of the external flooding events as documented in DCD Subsection 19.58.2.2 and Section 4.0 of APP-GW-GLR-101. No site specific external flood vulnerabilities have been identified and no further site specific PRA considerations are required.	Note 2

VEGP SUP 19.58-1

Table 19.58-201 (Sheet 4 of 10)
External Event Screening for VEGP

Category	Event	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency
Transportation and Nearby Facility Accidents	Aviation (commercial/ general/military)	N	Aircraft hazards are addressed in VEGP ESPA SSAR Subsection 3.5.1.6. All airports, airways, and military training routes, with the exception of commercial airway V185, were determined to be below the NRC Review Standard RS-002 screening threshold of 1E-07 for evaluating aircraft hazards. Due to the unavailability of traffic data for Airway V185, an evaluation was performed to calculate the maximum number of airway flights per year, above which the acceptance guideline of 1E-07 per year contained in RS-002 and NUREG-0800 are exceeded. The evaluation determined that approximately 51,100 flights per year would be required to reach the limiting crash probability of 1E-07. This value is higher than the total of all projected itinerant flights expected to utilize the airway. Therefore, based on the regulatory screening criteria and the airway traffic analysis, it can be concluded that the probability of a crash that would adversely impact VEGP Units 3 and 4 is less than 1.0E-07. This event frequency is bounded by the limiting value of 1.21E-06 events/year given in APP-GW-GLR-101.	<1.0E-07
	Marine (ship/barge)	N	As discussed in VEGP ESPA SSAR Subsection 2.2.3.1.3, there is no barge traffic past the VEGP site; therefore this event is not applicable to the VEGP site. Since the CDF given in APP-GW-GLR-101 Section 5.2 is based on the premise that a marine accident is a concern the CDF value given in APP-GW-GLR-101 is considered bounding.	Note 2

Table 19.58-201 (Sheet 5 of 10)
External Event Screening for VEGP

VEGP SUP 19.58-1

Category	Event	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency
Transportation and Nearby Facility Accidents (cont.)	Pipeline (gas/oil)	N	As discussed in VEGP ESPA SSAR Subsection 2.2.3.1.2, there are no natural gas pipelines within 10 miles of the VEGP site. No other pipelines carrying potentially hazardous materials are located within 5 miles of the VEGP site. APP-GW-GLR-101 evaluates a 30" gas pipeline approximately 1 mile from the AP1000 and concludes that the initiating event frequency for an event is expected to be less than 1E-07. Because the pipeline hazards at VEGP are well beyond this distance, it is concluded that the APP-GW-GLR-101 evaluation is bounding. Therefore, the potential for hazards from these sources are minimal and will not adversely affect the safe operation of VEGP Units 3 and 4.	Note 2
	Railroad	N	Potential explosion and flammable vapor cloud hazards to VEGP Units 3 and 4 resulting from railroad accidents are discussed in VEGP ESPA SSAR Subsection 2.2.3.1.4. The potential hazard resulting from railroad cars was evaluated using the methodology of RG 1.91. The maximum probable cargo based on RG 1.91 was used, along with a conservative TNT equivalency, which resulted in a safe standoff distance that was significantly less than the actual distance from the nearest railroad line to the site boundary (approximately 4.5 miles). Potential toxic hazards to control room habitability due to a release of hazardous chemicals resulting from a railcar accident are addressed in VEGP ESPA SSAR Subsection 2.2.3.2.1. This hazard was evaluated using the methodology of RG 1.78. The results of this evaluation concluded that no adverse impacts to VEGP Units 3 and 4 are expected. Based upon the quantitative consequence evaluations performed, no risk-important events related to railroad transportation have been identified for VEGP Units 3 and 4. Therefore, the evaluations presented in DCD Section 19.58 and APP-GW-GLR-101 are bounding to the VEGP Units 3 and 4 site.	Note 2

VEGP SUP 19.58-1

Table 19.58-201 (Sheet 6 of 10)
External Event Screening for VEGP

Category	Event	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency
Transportation and Nearby Facility Accidents (cont.)	Truck	N	Potential explosion and flammable vapor cloud hazards to VEGP Units 3 and 4 resulting from truck accidents are discussed in VEGP ESPA SSAR Subsection 2.2.3.1.1. The potential hazard resulting from trucks was evaluated using the methodology of RG 1.91. The maximum probable cargo based on RG 1.91 was used, along with a conservative TNT equivalency, which resulted in a safe standoff distance that was significantly less than the actual distance from the nearest highway to the site boundary (approximately 4.7 miles). Potential toxic hazards to control room habitability due to a release of hazardous chemicals resulting from a truck accident are addressed in VEGP ESPA SSAR Subsection 2.2.3.2.1. This hazard was evaluated using the methodology of RG 1.78. The results of this evaluation concluded that no adverse impacts to VEGP Units 3 and 4 are expected. Based upon the quantitative consequence evaluations performed, no risk-important events related to truck transportation have been identified for VEGP Units 3 and 4. Therefore, the evaluations presented in DCD Section 19.58 and APP-GW-GLR-101 are bounding to the VEGP Units 3 and 4 site.	Note 2

VEGP SUP 19.58-1

Table 19.58-201 (Sheet 7 of 10)
External Event Screening for VEGP

Category	Event	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency
Other events included in ESPA and COLA:	Major Depots and Storage Areas	N	Potential hazards from major depots and storage areas have been addressed in ESPA Subsection 2.2.3.2.2. The evaluation determined that the only potential hazard that required further evaluation was a postulated release of fuel oil from an accident spill at Plant Wilson. Based upon the evaluation performed it was determined that the postulated spill will not pose a toxicity hazard to VEGP Units 3 and 4. Note that the effect of an external fire at Plant Wilson is evaluated under the External Fire event. The evaluations performed for this external event meet the criteria in NRC Review Standard RS-002 and demonstrate through bounding analysis that the hazard does not adversely affect VEGP Units 3 and 4. Therefore, the hazard can be excluded from further consideration in the PRA analysis. This event is not specifically addressed in DCD Section 19.58 or in APP-GW-GLR-101. As discussed, the event screens out from further PRA considerations, therefore the evaluations presented in DCD Section 19.58 and APP-GW-GLR-101 are bounding to the VEGP Units 3 and 4 site.	Note 2

Table 19.58-201 (Sheet 8 of 10)
External Event Screening for VEGP

VEGP SUP 19.58-1

Category	Event	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency
Other events included in ESPA and COLA (cont.):	On-site Storage Tanks	N	Potential hazards from on-site storage tanks are addressed in ESPA Subsection 2.2.3.2.3 and COLA Subsection 2.2.3.2.3.1. Chemicals stored on site with low toxicity or volatility have been excluded from further consideration. Chemicals not excluded have been specifically evaluated. Chemicals with potential explosion or flammable vapor cloud hazards have been evaluated using the methodology of Regulatory Guide 1.91. Chemicals with potential hazards to control room personnel have been evaluated using the methodology of Regulatory Guide 1.78 and NUREG-0570. Based upon the quantitative evaluations performed, it is concluded that these evaluations demonstrate through bounding analyses that these hazards do not adversely affect VEGP Units 3 and 4. Therefore, the hazard can be excluded from further consideration in the PRA analysis. This event is not specifically addressed in DCD Section 19.58 or in APP-GW-GLR-101. As discussed, the event screens out from further PRA considerations, therefore the evaluations presented in DCD Section 19.58 and APP-GW-GLR-101 are bounding to the VEGP Units 3 and 4 site.	Note 2

VEGP SUP 19.58-1

Table 19.58-201 (Sheet 9 of 10)
External Event Screening for VEGP

Category	Event	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency
Other events included in ESPA and COLA (cont.):	External Fires	N	External fires have been addressed in ESPA Subsection 2.2.3.3 and in COLA Subsections 2.2.3.3.1 and 2.2.3.3.2. The external fire hazards evaluated included forest fires and fire due to an accident at an offsite industrial facility (Plant Wilson). Fire hazards related to on-site chemical storage and transportation accidents are evaluated separately under those specific hazard evaluations. The evaluations performed assessed heat flux, temperature rise and the effects of smoke (toxic chemicals) on control room personnel. Based on the above, it is demonstrated through bounding analysis that there are no external fire events that adversely affect VEGP Units 3 and 4. Therefore, no further consideration of external fires is required in the PRA analysis. This event is not specifically addressed in DCD Section 19.58 or in APP-GW-GLR-101, though AP1000 DCD Section 19.58 does state that the COL applicant should re-evaluate and include external fires in the site specific PRA if any site specific susceptibilities are found. As discussed above, no site specific susceptibilities have been identified for the VEGP Units 3 and 4 site, therefore the evaluations presented in DCD Section 19.58 and APP-GW-GLR-101 are bounding to the VEGP Units 3 and 4 site.	Note 2

VEGP SUP 19.58-1

Table 19.58-201 (Sheet 10 of 10)
External Event Screening for VEGP

Category	Event	Applicable to Site? (Y/N)[1]	Explanation of Applicability Evaluation	Event Frequency
Other events included in ESPA and COLA (cont.):	Radiological Hazards	N	An evaluation of potential radiological hazards to VEGP Units 3 and 4 from a postulated design basis accident in VEGP Unit 1 or 2 has been performed (ESPA Subsection 2.2.3.4) based on a LOCA in Unit 1 or 2, at uprated conditions, using the releases produced from the alternate source term (AST) methodology. The resultant dose from this analysis is comparable to the dose reported in DCD Tier 2, Table 15.6.5-3 for a postulated LOCA in the AP1000 and is less than the GDC 19 limits. As stated, the event is of equal or lesser damage potential than the dose reported for a LOCA in the AP1000, therefore the hazard can be excluded from further consideration in the PRA analysis. Note that this event is not specifically discussed in DCD Section 19.58 or in APP-GW-GLR-101. As discussed, the event screens out from further PRA considerations, therefore the evaluations presented in DCD Section 19.58 and APP-GW-GLR-101 are bounding to the VEGP Units 3 and 4 site.	Note 2

Notes:

1. An event is applicable (Y) to the VEGP site if the initiating event frequency is greater than 1E-07, or if a quantitative consequence evaluation has demonstrated that there are site specific parameters that exceed the parameters used APP-GW-GLR-101. An event is not applicable (N) to the VEGP site if the initiating event frequency is less than 1E-07 or if the quantitative consequence evaluation performed in the FSAR/SSAR has demonstrated that the event will not adversely impact the safe operation of VEGP Units 3 and 4.

2. A specific event frequency for this event has not been determined. A deterministic quantitative consequence evaluation has been performed that has demonstrated that the event does not adversely impact the safe operation of VEGP Units 3 & 4. Additional details are provided in the "Explanation of Applicability Evaluation" along with references to the applicable FSAR/SSAR Subsections.

19.59 PRA RESULTS AND INSIGHTS

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

19.59.10.5 Combined License Information

STD COL 19.59.10-1
STD COL 19.59.10-6

A review of the differences between the as-built plant and the design used as the basis for the AP1000 seismic margins analysis will be completed prior to fuel load. A verification walkdown will be performed with the purpose of identifying differences between the as-built plant and the design. Any differences will be evaluated and the seismic margins analysis modified as necessary to account for the plant-specific design, and any design changes or departures from the certified design. A comparison of the as-built SSC high confidence, low probability of failures (HCLPFs) to those assumed in the AP1000 seismic margin evaluation will be performed prior to fuel load. Deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis will be evaluated to determine if vulnerabilities have been introduced. The requirements to which the equipment is to be purchased are included in the equipment specifications. Specifically, the equipment specifications include:

1. Specific minimum seismic requirements consistent with those used to define the AP1000 DCD Table 19.55-1 HCLPF values.

   This includes the known frequency range used to define the HCLPF by comparing the required response spectrum (RRS) and test response spectrum (TRS). The test response spectra are chosen so as to demonstrate that no more than one percent rate of failure is expected when the equipment is subjected to the applicable seismic margin ground motion for the equipment identified to be applicable in the seismic margin insights of the site-specific PRA.The range of frequency response that is required for the equipment with its structural support is defined.

2. Hardware enhancements that were determined in previous test programs and/or analysis programs will be implemented.

STD COL 19.59.10-2

A review of the differences between the as-built plant and the design used as the basis for the AP1000 PRA and DCD Table 19.59-18 will be completed prior to fuel load. The plant-specific PRA-based insight differences will be evaluated and the plant-specific PRA model modified as necessary to account for plant-specific design and any design changes or departures from the design certification PRA.

As discussed in Section 19.58.3, it has been confirmed that the Winds, Floods and Other External Events analysis documented in DCD Section 19.58 is applicable to the site. The site-specific design has been evaluated and is consistent with the AP1000 PRA assumptions. Therefore, Section 19.58 of the AP1000 DCD is applicable to this design.

STD COL 19.59.10-3

A review of the differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analyses will be completed prior to fuel load. Plant specific internal fire and internal flood analyses will be evaluated and the analyses modified as necessary to account for the plant-specific design, and any design changes or departures from the certified design.

STD COL 19.59.10-4

The AP1000 Severe Accident Management Guidance (SAMG) from APP-GW-GLR-070, Reference 1 of DCD Section 19.59, is implemented on a site-specific basis. Key elements of the implementation include:

- SAMG based on APP-GW-GLR-070 is provided to Emergency Response Organization (ERO) personnel in assessing plant damage, planning and prioritizing response actions and implementing strategies that delineate actions inside and outside the control room.
- Severe accident management strategies and guidance are interfaced with the Emergency Operating Procedures (EOP's) and Emergency Plan.
- Responsibilities for authorizing and implementing accident management strategies are delineated as part of the Emergency Plan.
- SAMG training is provided for ERO personnel commensurate with their responsibilities defined in the Emergency Plan.

STD COL 19.59.10-5

A thermal lag assessment of the as-built equipment required to mitigate severe accidents (hydrogen igniters and containment penetrations) will be performed to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. This assessment will be performed prior to fuel load and is required only for equipment used for severe accident mitigation that has not been tested at severe accident conditions. The ability of the as-built equipment to perform during severe accident hydrogen burns will be assessed using the Environment Enveloping method or the Test Based Thermal Analysis method discussed in EPRI NP-4354 (DCD Section 19.59, Reference 3).

STD COL 19.59.10-6
VEGP COL 19.59.10-6

As discussed in Subsection 19.55.6.3, it has been confirmed that the Seismic Margin Analysis (SMA) documented in DCD Section 19.55 is applicable to the site. The site-specific effects (i.e., soil-related failure modes, etc.) have been evaluated and it was concluded that the plant-specific plant-level HCLPF value is equal to or greater than 1.67 times the site-specific GMRS peak ground acceleration.

Add the following new information after DCD Subsection 19.59.10.5:

STD SUP 19.59-1

19.59.10.6 PRA Configuration Controls

PRA configuration controls contain the following key elements:

- A process for monitoring PRA inputs and collecting new information.
- A process that maintains and updates the PRA to be reasonably consistent with the as-built, as operated plant.
- A process that considers the cumulative impact of pending changes when applying the PRA.
- A process that evaluates the impact of changes on currently implemented risk-informed decisions that have used the PRA.
- A process that maintains configuration control of computer codes used to support PRA quantification.
- A process for upgrading the PRA to meet PRA standards that the NRC has endorsed.
- Documentation of the PRA.

PRA configuration controls are consistent with the regulatory positions on maintenance and upgrades in Regulatory Guide 1.200.

<u>Schedule for Maintenance and Upgrades of the PRA</u>

The PRA update process is a means to reasonably reflect the as designed and as operated plant configurations in the PRA models. The PRA upgrade process includes an update of the PRA plus a general review of the entire PRA model, and as applicable the application of new software that implements a different methodology, implementation of new modeling techniques, as well as a comprehensive documentation effort.

- During construction, the PRA is upgraded prior to fuel load to cover those initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to the scheduled date of the initial fuel load for a Level 1 and Level 2 PRA.

- Prior to license renewal the PRA is upgraded to include all modes of operation.
- During operation, PRA updates are completed as part of the upgrade process at least once every four years.
- A screening process is used to determine whether a PRA update should be performed more frequently based upon the nature of the changes in design or procedures. The screening process considers whether the changes affect the PRA insights. Changes that do not meet the threshold for immediate update are tracked for the next regulatory scheduled update. If the screening process determines that the changes do warrant a PRA update, the update is made as soon as practicable consistent with the required change importance and the applications being used.

PRA upgrades are performed in accordance with 10 CFR 50.71(h).

Process for Maintenance and Upgrades of the PRA

Various information sources are monitored to determine changes or new information that affects the model assumptions or quantification. Plant specific design, procedure, and operational changes are reviewed for risk impact. Information sources include applicable operating experience, plant modifications, engineering calculation revisions, procedure changes, industry studies, and NRC information.

The PRA upgrade includes initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect one year prior to each required upgrade.

This PRA maintenance and update incorporates the appropriate new information including significant modeling errors discovered during routine use of the PRA.

Once the PRA model elements requiring change are identified, the PRA computer models are modified and appropriate documents revised. Documentation of modifications to the PRA model include the changes as well as the upgraded portions clearly indicating what has been changed. The impact on the risk insights is clearly indicated.

PRA Quality Assurance

Maintenance and upgrades of the PRA are subject to the following quality assurance provisions:

Procedures identify the qualifications of personnel who perform the maintenance and upgrade of the PRA.

Procedures provide for the control of PRA documentation, including revisions.

For updates of the PRA, procedures provide for independent review, or checking of the calculations and information.

Procedures provide for an independent review of the model after an upgrade is completed. Additionally, after the PRA is upgraded, the PRA is reviewed by outside PRA experts such as industry peer review teams and the comments incorporated to maintain the PRA current with industry practices. Peer review findings are entered into a tracking system. PRA upgrades receive a peer review for those aspects of the PRA that are upgraded.

PRA models and applications are documented in a manner that facilitates peer review as well as future updates and applications of the PRA by describing the processes that were used, and provide details of the assumptions made and their bases. PRA documentation is developed such that traceability and reproducibility is maintained. PRA documentation is maintained in accordance with Regulatory Position 1.3 of Regulatory Guide 1.200.

Procedures provide for appropriate attention or corrective actions if assumptions, analyses, or information used previously are changed or determined to be in error. Potential impacts to the PRA model (i.e., design change notices, calculations, and procedure changes) are tracked. Errors found in the PRA model between periodic updates are tracked using the site tracking system.

PRA-Related Input to Other Programs and Processes

The PRA provides input to various programs and processes, such as the Maintenance Rule implementation, reactor oversight process, the RAP, and the RTNSS program. The use of the PRA in these programs is discussed below, or cross-references to the appropriate FSAR sections are provided.

PRA Input to Design Programs and Processes

The PRA insights identified during the design development are discussed in DCD Subsection 19.59.10.4 and summarized in DCD Table 19.59-18.
DCD Section 14.3 summarizes the design material contained in AP1000 that has been incorporated into the Tier 1 information from the PRA. A discussion of the plant features important to reducing risk is provided in DCD Subsection 19.59.9.

PRA Input to the Maintenance Rule Implementation

The PRA is used as an input in determining the safety significance classification and bases of in-scope SSCs. SSCs identified as risk-significant via the Reliability

Assurance Program for the design phase (DRAP, Section 17.4) are included within the initial Maintenance Rule scope as high safety significance SSCs.

For risk-significant SSCs identified via DRAP, performance criteria are established, by the Maintenance Rule expert panel using input from the reliability and availability assumptions used in the PRA, to monitor the effectiveness of the maintenance performed on the SSCs.

The Maintenance Rule implementation is discussed in Section 17.6.

PRA Input to the Reactor Oversight Process

The mitigating systems performance indicators (MSPI) are evaluated based on the indicators and methodologies defined in NEI 99-02 (Reference 201).

The Significance Determination Process (SDP) uses risk insights, where appropriate, to determine the safety significance of inspection findings.

PRA Input to the Reliability Assurance Program

The PRA input to the Reliability Assurance Program is discussed in DCD Subsection 19.59.10.1.

PRA Input to the Regulatory Treatment of Nonsafety-Related Systems Programs

The importance of nonsafety-related SSCs in the AP1000 has been evaluated using PRA insights to identify SSCs that are important in protecting the utility's investment and for preventing and mitigating severe accidents. These investment protection systems, structures and components are included in the D-RAP/MR Program (refer to Section 17.4), which provides confidence that availability and reliability are designed into the plant and that availability and reliability are maintained throughout plant life through the maintenance rule. Technical Specifications are not required for these SSCs because they do not meet the selection criteria applied to the AP1000 (refer to Subsection 16.1.1).

MOV Program

The MOV Program includes provisions to accommodate the use of risk-informed inservice testing of MOVs (Subsection 3.9.6).

19.59.11 REFERENCES

Add the following to the end of DCD Subsection 19.59.11.

201. NEI 99-02 Nuclear Energy Institute, “Regulatory Assessment Performance Indicator Guideline,” Technical Report NEI 99-02, Revision 5, July 2007.

APPENDIX 19A THERMAL HYDRAULIC ANALYSIS TO SUPPORT SUCCESS CRITERIA

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19B EX-VESSEL SEVERE ACCIDENT PHENOMENA

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19C ADDITIONAL ASSESSMENT OF AP1000 DESIGN FEATURES

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19D EQUIPMENT SURVIVABILITY ASSESSMENT

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19E SHUTDOWN EVALUATION

This section of the referenced DCD is incorporated by reference with no departures or supplements.

APPENDIX 19F MALEVOLENT AIRCRAFT IMPACT

This section of the referenced DCD is incorporated by reference with no departures or supplements.