Donald J. Trump Generating Plant, Units 1-4

COL Application

Part 2

Final Safety Analysis Report

REVISION 1

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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.

FA 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

FA COL 1.1-1 Add the following text at the beginning of DCD Subsection 1.1.1:

The Project Matador site in Carson County lies approximately 17 miles northeast of Amarillo, Texas, immediately south of the DOE's Pantex Plant. The site is under institutional control via a 99-year lease with Texas Tech University and has undergone extensive environmental and geotechnical study, including soil borings, aquifer assessments, and cultural resource surveys.

The site is bounded by State Highway 60 and lies within proximity to critical natural gas infrastructure, including intrastate and interstate pipelines. The regional terrain is flat with shallow playa basins and underlain by the Ogallala and Dockum Aquifers. The area is not within a designated FEMA floodplain, and no active fault lines intersect the immediate region. These conditions satisfy NRC criteria for safe site suitability as defined under 10 CFR Part 100.

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:

FA COL 1.1-1 Table 1.1-203 displays the anticipated schedule for construction and operation of four AP1000 units at the Project Matador 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.

FA SUP 1.1-5 Table 1.1-201 provides a list of acronyms used in the DJTGP Units 1-4 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.

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

FA 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

FA SUP 1.1-5

Table 1.1-201 (Sheet 2 of 6)
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

FA SUP 1.1-5

Table 1.1-201 (Sheet 3 of 6)
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

FA SUP 1.1-5

Table 1.1-201 (Sheet 4 of 6)
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

FA SUP 1.1-5

Table 1.1-201 (Sheet 5 of 6)
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

FA SUP 1.1-5

Table 1.1-201 (Sheet 6 of 6)
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.

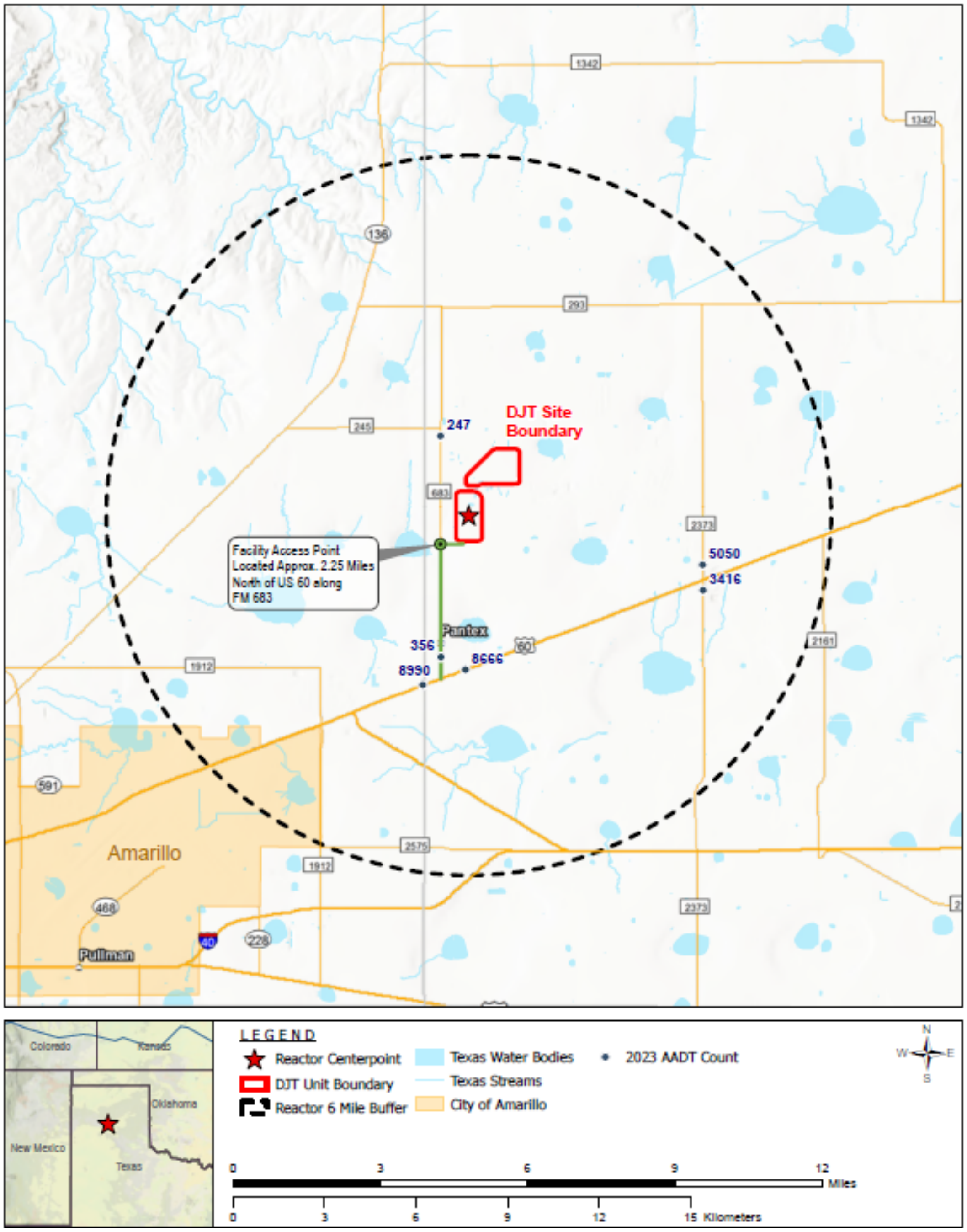
VEGP SUP 1.1-8

FA SUP 1.1-6

Table 1.1-203
Anticipated Schedule for Construction and Operation of the first AP1000 Unit at FA Site

Activity	Start(1)	Finish(1)	Duration
	UNIT 1		
Early Procurement Activities	4th Q 2025	--	--
Site Preparation	1st Q 2026	3rd Q 2028	24 Months
Commence Construction (Safety-related activities)	2nd Q 2027 (LWA)		--
Fuel Load, Commence Start-Up	2nd Q 2032	4th Q 2032	6 Months
Commercial Operation	December 2032		
	UNIT 2-3-4 – TO BE PROVIDED LATER		

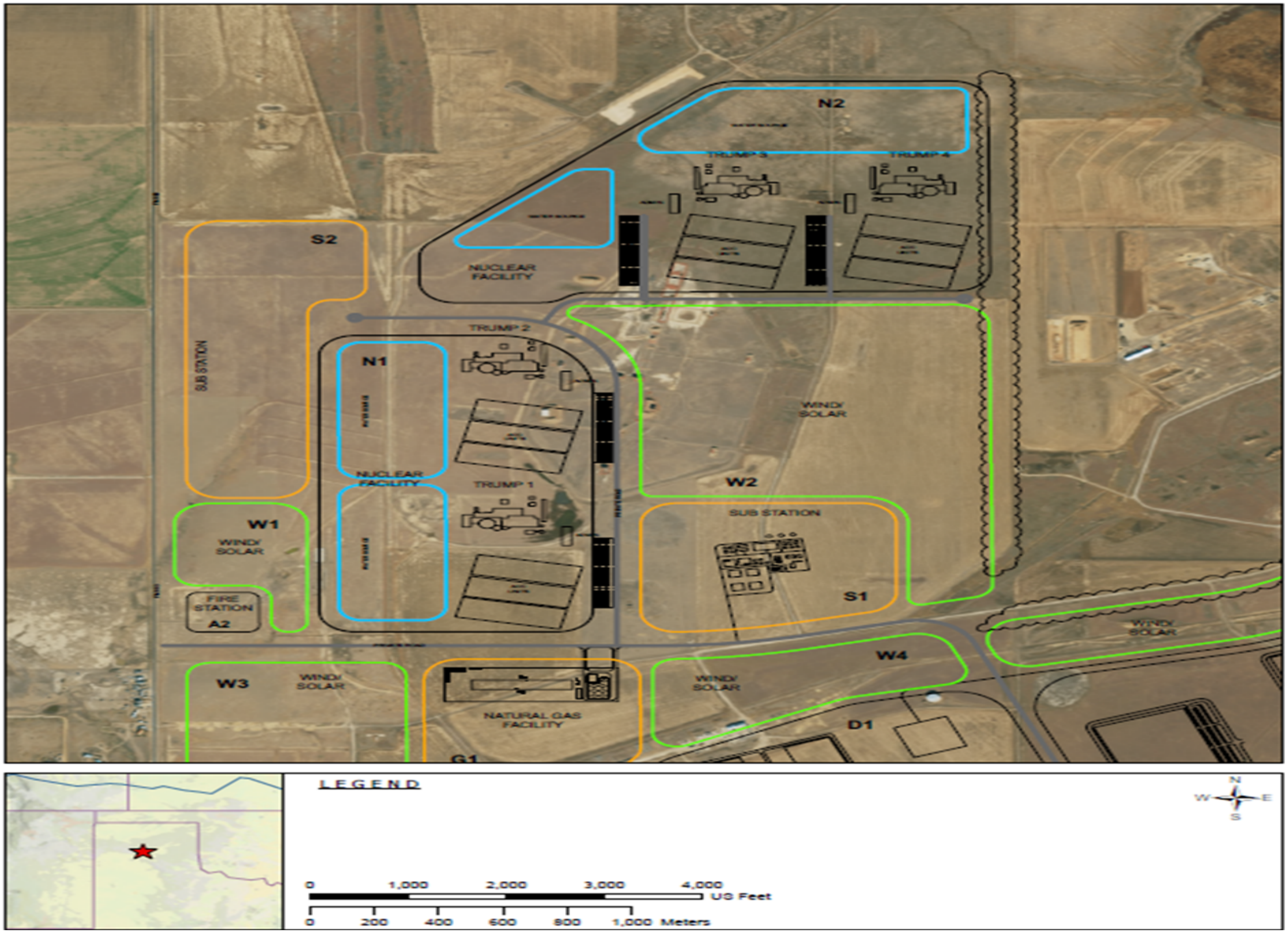
(1) Activities that do not indicate a start or finish date are milestones, and represent the anticipated commencement (start) or completion (finish) of the activity.



FA COL 2.1-1

Figure 1.1-201

Site Location Map


N2
TRUMP 3
TRUMP 4
S2
NUCLEAR FACILITY
TRUMP 2
SUB STATION
N1
WIND/ SOLAR
NUCLEAR FACILITY
TRUMP 1
W2
SUB STATION
W1
WIND/ SOLAR
FIRE STATION
A2
S1
WIND/ SOLAR
W4
W3
WIND/ SOLAR
NATURAL GAS FACILITY
WIND/ SOLAR
D1
G1
LEGEND
N
W
E
S
0
1,000
2,000
3,000
4,000
US Feet
0
200
400
600
800
1,000
Meters

1.2 GENERAL PLANT DESCRIPTION

TO BE PROVIDED LATER – Fermi America site-specific information under development.

1.2.2 SITE DESCRIPTION

TO BE PROVIDED LATER – Fermi America site-specific information under development.

.

1.2.3 PLANT ARRANGEMENT DESCRIPTION

TO BE PROVIDED LATER – Fermi America site-specific information under development.

.

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:

FA SUP 1.4-1 Fermi America LLC is a vertically integrated advanced energy and infrastructure development company established for the design, licensing, financing, construction, and operation of nuclear, natural gas, and solar generation assets. The company serves as the master developer of **Project Matador – The President Donald J. Trump Advanced Energy and Intelligence Campus**, located in Carson County, Texas.

The company's core business model is centered on the deployment of AP1000 nuclear generation units integrated with natural gas and renewable generation sources under a sovereign-controlled leasehold. Fermi America specializes in:

- Providing energy for hyperscale tenants;
- Deployment of advanced cooling technologies, including air-cooled condensers;
- Turnkey digital and energy platform development for defense-aligned and AI-intensive data center operations;
- Full lifecycle nuclear facility management, from licensing through decommissioning.

Fermi America is structured to oversee site licensing, regulatory compliance, infrastructure construction, tenant power delivery, and operational integrity. Its leadership team includes nuclear energy veterans, infrastructure developers, financial architects, and digital systems engineers.

Future affiliates or subsidiaries—such as **Fermi Nuclear Operations LLC** or **Fermi Energy Partners LLC**—may be formed to execute specialized operations (e.g., operator services, tenant interfacing, or REIT structuring) under the governance of the parent LLC. All such entities will remain under the direct oversight of Fermi America and fully accountable to NRC requirements and license conditions.

FA SUP 1.4-2

1.4.2.8 Other Contractors

TO BE PROVIDED LATER – Fermi America site-specific information under development

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1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

This section of the referenced DCD is incorporated by reference with no departures 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 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)

STD SUP 1.6-1

Table 1.6-201
Additional Material Referenced

	Author/ Report Number[a]	Title	Revision	FSAR Section	Document Transmittal	ADAMS Accession Number
VEGP SUP 1.6-2	VEGP ESP A 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.

VEGP SUP 1.6-2
(Unless Otherwise Noted)

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

	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.
	1.2	General Site Description	TO BE PROVIDED LATER.
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	TO BE PROVIDED LATER
	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

VEGP SUP 1.6-2
(Unless Otherwise Noted)

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

	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	TO BE PROVIDED LATER
VEGP ESP VAR 2.2-1	2.2	Identification of Potential Hazards in Site Vicinity	TO BE PROVIDED LATER.
	2.3	Meteorology	TO BE PROVIDED LATER
	2.4	Hydrologic Engineering	TO BE PROVIDED LATER
	2.5	Geology, Seismology, and Geotechnical Engineering	TO BE PROVIDED LATER
	3.5.1.6	Aircraft Hazards	Section 3.5.1.6, Aircraft Hazards

VEGP SUP 1.6-2
(Unless Otherwise Noted)

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

	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
	13.3	Emergency Planning	TO BE PROVIDED LATER
	13.6	Industrial Security	TO BE PROVIDED LATER
	13.7	Fitness for Duty	Section 13.7, Fitness for Duty
	15	Accident Analyses	TO BE PROVIDED LATER.

VEGP SUP 1.6-2
(Unless Otherwise Noted)

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

SSAR Section	SSAR Section Title	Corresponding FSAR Section
17	Quality Assurance	TO BE PROVIDED LATER.

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	TO BE PROVIDED LATER	8.2	8.2-201
ZBS	TO BE PROVIDED LATER	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 identified 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.

Demonstrations that the DJT Units 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 - TO BE PROVIDED LATER.

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

VEGP SUP 1.8-1

Table 1.8-201 (Sheet 2 of 2)
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

VEGP SUP 1.8-2

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

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

VEGP SUP 1.8-2

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

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

VEGP SUP 1.8-2

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

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

VEGP SUP 1.8-2

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

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

VEGP SUP 1.8-2

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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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

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

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

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

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

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

NO.	ESP PERMIT CONDITION	COLA LOCATION
1	TO BE PROVIDED LATER.	

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
2	TO BE PROVIDED LATER.	FSAR Subsection 13.3.8 Part 10, License Condition 4

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

VEGP SUP 1.8-6

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

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

VEGP SUP 1.8-6

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

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

VEGP SUP 1.8-6

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

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

VEGP SUP 1.8-6

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

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

VEGP SUP 1.8-6

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

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

VEGP SUP 1.8-6

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

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. TO BE PROVIDED LATER

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 TO BE PROVIDED LATER.

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.

STD COL 1.9-1
(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

STD COL 1.9-1
(Unless Otherwise Noted)

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)

STD COL 1.9-1
(Unless Otherwise Noted)

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

STD COL 1.9-1
(Unless Otherwise Noted)

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	1.68	Initial Test Program for Water-Cooled Nuclear Power Plants (Rev. 3, March 2007)	Note b Note b Table 2.0-201
	1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition) (Rev. 3, November 1978)	DCD discussion only; see DCD Table 1.9-1
	1.71	Welder Qualification for Areas of Limited Accessibility (Rev 1, 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	1.75	Criteria for Independence of Electrical Safety Systems (Rev 3, February 2005)	Note b 1.1.6.1
	1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (Rev. 1, March 2007)	DCD discussion only; see DCD Table 1.9-1
	1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (PR-1, January 2006)	DCD discussion only; see DCD Table 1.9-1
	1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev 0, May 1974)	Table 2.0-201
VEGP COL 1.9-1			Note b
			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 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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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]
	1.165	Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion (Rev. 0, March 1997)	Not referenced; see Appendix 1AA
	1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post Earthquake Actions (Rev. 0, March 1997)	6.2.5.1 6.2.5.2.2 16 (TS 5.5.8)
VEGP COL 1.9-1	1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event (Rev. 0, March 1997)	Note b
	1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 1, February 2004)	3.7.4.4
	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)	3.7.4.4
	1.175	An Approach for Plant-Specific, Risk-Informed Decision making: Inservice Testing (Rev. 0, August 1998)	DCD discussion only; see DCD Table 1.9-1
	1.177	An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications (Rev. 0, August 1998)	Not referenced; see Appendix 1AA
			Not referenced; see Appendix 1AA
			16 (TS Bases 3.5.1) 16 (TS Bases 3.7.10)

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		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]
		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		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)
			Not referenced; see Appendix 1AA
	1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown (Rev. 0, May 2001)	3.9.6.3
	1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code (Rev. 0, June 2003)	Not referenced; see Appendix 1AA
	1.193	ASME Code Cases Not Approved for Use (Rev 1, August 2005)	2.3.4.3
	1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (Rev. 0, June 2003)	Not referenced; see Appendix 1AA
	1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (Rev. 0, May 2003)	6.4.3
	1.196	Control Room Habitability at Light-Water Nuclear Power Reactors (Rev. 1, January 2007)	DCD discussion only; see DCD Table 1.9-1
	1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (Rev. 0, May 2003)	

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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 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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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 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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		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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Conformance with SRP Acceptance Criteria

	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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Conformance with SRP Acceptance Criteria

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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	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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	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.
	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.
13.1.1 –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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	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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		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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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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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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	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.

STD SUP 1.9-1
(Unless Otherwise Noted)

Table 1.9-202 (Sheet 26 of 27)[a]
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.

STD SUP 1.9-1
(Unless Otherwise Noted)

Table 1.9-202 (Sheet 27 of 27)[a]
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.

VEGP SUP 1.9-2 h) Conformance with RS-002 and NUREG-0800 criteria contained in referenced ESPA. Refer to ESPA SSAR Table 1-2.

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Table 1.9-203 (Sheet 1 of 15)
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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Table 1.9-203 (Sheet 2 of 15)
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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Table 1.9-203 (Sheet 4 of 15)
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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Table 1.9-203 (Sheet 5 of 15)
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

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Table 1.9-203 (Sheet 7 of 15)
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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Table 1.9-203 (Sheet 8 of 15)
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

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Table 1.9-203 (Sheet 10 of 15)
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

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Table 1.9-203 (Sheet 11 of 15)
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

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Table 1.9-203 (Sheet 12 of 15)
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

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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

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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

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(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
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

VEGP COL 1.9-2 (margin note beside 82-12)

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(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

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(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 Minimum separation between Units TO BE PROVIDED LATER.

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). Applicability TO BE PROVIDED LATER.

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.

STD SUP 1.10-1 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.

STD SUP 1.10-1

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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(Unless Otherwise Noted)

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.

STD COL 1.9-1 (Unless Otherwise Noted)

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

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

STD COL 1.9-1
(Unless Otherwise Noted)

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	

STD COL 1.9-1 (Unless Otherwise Noted)	Criteria Section	Referenced Criteria	FSAR Position	Clarification/ Summary Description of Exceptions
VEGP COL 1.9-1	**Regulatory Guide 1.70, Rev. 3, 11/78, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)**			
	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.			
VEGP COL 1.9-1	**Regulatory Guide 1.76, Rev. 1, 3/07 – Design-Basis Tornado and Tornado Missiles for 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.			
			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.			
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 1.97, 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.			

STD COL 1.9-1 (Unless Otherwise Noted)	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	

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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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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.

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

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

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

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

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

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

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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.

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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.

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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	

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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."

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.

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

FA SUP 2.0-1 Chapter 2 describes the characteristics and site-related design parameters of the Donald J. Trump Generating Plant Units 1 through 4 (DJTGP). 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 DJTGP 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 DJTGP. 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.

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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

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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

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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

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Table 2.0-201 (Sheet 4 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
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:			

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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:			

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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)		

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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).	ESPA SSAR Table 1-1	Yes
		Maximum local PMP flood elevation is 219.47 ft MSL, which is 0.53 feet below plant elevation (220 ft MSL).	Subsection 2.4.2	

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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

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Table 2.0-201 (Sheet 9 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
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.

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

FA SUP XXX

2.1.4.1 SITE LOCATION AND GEOGRAPHIC SETTING

The Fermi America facility, referred to as Project Matador, is located in Carson County, Texas, approximately 17 miles northeast of downtown Amarillo. The geographic coordinates of the proposed Nuclear Island centerpoint are approximately 35.3987° N latitude and 101.2495° W longitude, with the site situated at an average elevation of 3,500 feet above sea level. The 5,855-acre parcel is governed under a 99-year sovereign lease agreement with Texas Tech University, a constitutionally authorized public university system of the State of Texas.

The site lies to the south and west of U.S. Department of Energy's Pantex Plant—a facility dedicated to nuclear materials assembly, storage, and national security operations. This proximity provides an additional layer of physical and institutional buffer from high-density population areas. Access to the site is available via State Highway 60 and Farm-to-Market Road 2373, which intersects the southwestern perimeter of the leased area. The surrounding land is sparsely developed, consisting primarily of dryland agriculture, natural gas infrastructure, and ranching operations.

Topographically, the site is located on the Southern High Plains—a flat to gently rolling plateau dominated by loessal soils and shallow drainage features. The project site is not in a FEMA-designated flood zone, and no major surface water bodies traverse the exclusion boundary. Terrain data, topographic maps, and the U.S. Geological Survey (USGS) Amarillo East Quadrangle confirm that the site is free of significant seismic, landslide, or erosion hazards. A complete map package, including the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and site boundary, is provided in Appendix 2.

FA SUP XXX

2.1.4.2 POPULATION DISTRIBUTION AND PROJECTIONS

The site is located in a low-population density region of the Texas Panhandle. Based on U.S. Census Bureau 2020 data and regional projections through 2045, the population within a 10-mile radius of the reactor site is estimated to be under 1,800 permanent residents, the majority of whom reside in small, unincorporated communities such as Washburn, Panhandle, and Conway. Within a 50-mile radius, the population reaches approximately 275,000, primarily concentrated in Amarillo and the outskirts of Potter and Randall Counties.

Transient populations are limited. The Pantex Plant maintains a stable workforce with tightly controlled site access and negligible residential or visitor turnover. The Fermi America project anticipates a peak construction workforce of up to 9,000, with a smaller long-term operations workforce of 1,200–1,800 employees. These figures will be factored into transient dose pathway and emergency response planning, as described in Part 5 of this COLA. Projected population growth for the region remains moderate, with estimated increases of less than 1.2% annually over the 20-year horizon, based on Texas Demographic Center forecasts. This trend supports the continued suitability of the site from a radiological safety and emergency planning standpoint.

FA SUP XXX

2.1.4.3 EXCLUSION AREA AND CONTROL

Fermi America, through its sovereign lease from Texas Tech University, retains exclusive legal and operational control over all activities within the defined Exclusion Area Boundary (EAB). The EAB has been established at a radius of 1,850 meters (approximately 1.15 miles) from the Nuclear Island centerline, consistent with the guidance of 10 CFR 100.21(c) and validated using AP1000 emergency planning and radiological dispersion modeling.

The exclusion area lies entirely within the leased boundary and is subject to enforceable institutional control provisions that allow Fermi America to prevent, restrict, or remove unauthorized public access or incompatible land uses. All construction, security, and environmental monitoring activities within the EAB are under Fermi America's operational jurisdiction, and no residential, commercial, or public activities are permitted without prior review and formal authorization.

The Low Population Zone (LPZ) extends outward to a distance of 5,000 meters (approximately 3.1 miles), encompassing only rural land uses, industrial rights-of-way, and federally controlled buffer space associated with the Pantex Plant. As such, the LPZ meets NRC criteria for accident consequence mitigation and emergency planning, as validated by the dose evaluations presented in FSAR Chapter 15 and the radiological impact assessments included in Part 3 of the COLA.

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.

FA SUP 2.2-1

2.2.4.1 NEARBY INDUSTRIAL FACILITIES

The region surrounding the site is rural and characterized by agricultural and energy sector land use. The primary industrial facilities in proximity to the site include:

- **Pantex Plant (DOE/NNSA)**: Located approximately 5.5 miles northeast of the reactor centerline, the Pantex Plant serves as a secure nuclear materials handling and assembly facility. All explosive operations at Pantex are conducted within secure perimeters and hardened structures. No credible accident scenarios have been identified that would result in explosive overpressure, toxic dispersion, or radiological release affecting the Fermi America exclusion area or safety-related SSCs.
- **Gas Compression and Gathering Stations**: Several natural gas compression facilities, pipeline valves, and pigging stations are located within a 10-mile radius. These facilities are governed by Texas Railroad Commission and PHMSA safety standards and do not contain significant inventories of hazardous chemicals in excess of screening thresholds outlined in RG 1.91. Terracon’s site hazard survey confirms that no single-point failure, explosion, or gas release scenario from these stations could produce overpressure exceeding 1 psi at the Fermi America EAB.
- **Wind Energy Infrastructure**: Wind turbines and associated control stations are distributed intermittently across the county but are located several miles away from the planned reactor footprint and pose no credible impact to nuclear safety or plant operations.

No chemical manufacturing, refinery, fertilizer, or large-scale hazardous materials processing facilities are located within 10 miles of the site. Industrial activity within the LPZ is limited to agricultural operations, small equipment repair shops, and utilities infrastructure.

FA SUP 2.2-2

2.2.4.2 TRANSPORTATION ROUTES AND PIPELINES

Transportation routes within the vicinity of the Fermi America site are limited in scale and traffic density.

- **Highways**: State Highway 60 lies approximately 1.5 miles southwest of the site and carries modest commercial and private traffic. Farm-to-Market Roads 2373 and 293 parallel the site but do not intersect the EAB. Historical accident data from TxDOT indicates low incidence of hazardous material transport events within this corridor.
- **Railroads**: A BNSF freight line runs parallel to SH-60, over 1.2 miles from the site boundary. While it does support occasional transport of petroleum-based goods, the distance, shielding topography, and prevailing wind direction significantly reduce any credible risk from toxic vapor clouds or fireball overpressure reaching safety-critical structures.
- **Pipelines**: The site is situated near multiple natural gas pipelines including interstate and intrastate corridors. All pipelines are located more than 1,200 feet from the EAB and are designed and maintained under PHMSA pressure integrity requirements. Modeled rupture scenarios confirm that overpressure at the plant boundary would not exceed the NRC’s 1 psi threshold for safety assessment.

- **Airports**: Rick Husband Amarillo International Airport is located approximately 22 miles southwest of the site. The site is not under any commercial flight path, as the airspace overhead is restricted to commercial traffic (FAA control area P-47). The risk of aircraft impact is addressed in Chapter 3 and determined to be below NRC threshold concern based on FAA traffic density and flight corridor analysis.
- **Waterways**: There are no navigable waterways, commercial ports, or barge operations in proximity to the site. The site is entirely inland and not subject to fluvial or coastal shipping hazards.

FA SUP 2.2-3

2.2.4.3 MILITARY FACILITIES

The only military-affiliated facility near the Fermi America site is the Department of Energy's Pantex Plant, which operates under the jurisdiction of the National Nuclear Security Administration (NNSA). While Pantex conducts national security-related operations involving nuclear materials, it does not house aircraft, live-fire training, or ordnance testing ranges that could affect plant safety. Its function as a secure buffer zone indirectly enhances the strategic siting profile of the Fermi America project.

There are no active military bases, airfields, or testing ranges within 25 miles of the site. The nearest major military installation is Cannon Air Force Base, located over 80 miles away in Clovis, New Mexico, which does not conduct low-altitude training exercises over the Carson County area.

FA SUP 2.2-4

2.2.5 HAZARD ANALYSIS AND SCREENING

FermiAmerica conducted a comprehensive hazard screening analysis in alignment with NRC Regulatory Guide 1.91 and NEI 10-06. The review used conservative modeling assumptions to assess whether any external hazard—such as vapor cloud ignition, pipeline rupture, chemical tank failure, or transportation accident—could generate an overpressure, thermal flux, or chemical dose exceeding NRC thresholds at the Exclusion Area Boundary (EAB).

No nearby facility or transportation corridor was found to pose a credible hazard to the safe operation of the AP1000 units. For any future identified source, bounding analyses will demonstrate:

- Explosion overpressure at the EAB is <1 psi;
- Toxic chemical concentrations remain below ERPG-2 and TEEL-2 thresholds;
- Flammable vapor clouds would dissipate prior to impacting safety-related areas;
- Fireball radiant energy does not exceed 1,800 BTU/ft² over 0.5 seconds.

These conclusions will be supported by modeling using ALOHA, PHAST, and EPA's RMP*Comp platforms and validated against AP1000 hardening criteria and spatial separations.

2.3 METEOROLOGY

This section of the referenced DCD is incorporated by reference with the following departures 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

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

2.3.7 REFERENCES

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

FA SUP 2.3-1

2.3.8 FERMI SPECIFIC SUPPLIMENTAL METORLOGIC INFORMATION

The following information is provided for COLA review based on publicly available information regarding meteorologic features at the project matador site. Much of this information will likely replace ESP related information from the Vogtle 3&4 COLA application, and will be refined prior to formal submittal. TO BE PROVIDED LATER.

FA SUP 2.3-2

2.3.8.1 REGIONAL CLIMATE

The Project matador site in Carson County, Texas is situated in the Southern High Plains and is characterized by a semi-arid climate typical of the western Texas Panhandle. The regional climate features large diurnal and seasonal temperature variations, low annual precipitation, persistent wind regimes, and generally clear sky conditions.

Average daily high temperatures during the summer months exceed 90°F (32°C), while winter temperatures frequently fall below freezing. The recorded historical temperature range extends from approximately -10°F (-23°C) to 110°F (43°C). Relative humidity values range from 25% to 45% annually, with occasional surges above 80% during spring and early summer storm systems. Annual precipitation averages between 17 to 21 inches, the majority falling between May and September. Snowfall is relatively infrequent, averaging 15 inches annually, and does not typically result in long-term ground cover or access impairments.

The regional wind regime is dominated by southerly and westerly winds, with an average sustained speed of 12 to 15 mph and seasonal gusts regularly exceeding 30 mph. This pattern, combined with the region's flat terrain, enhances atmospheric mixing and supports effective radiological and chemical dispersion in the event of an accidental release.

FA SUP 2.3-3

2.3.8.2 LOCAL METEORLOGICAL MONITORING AND SITE CONDITIONS

Fermi America will establish an on-site meteorological monitoring system in accordance with the guidance of NRC Regulatory Guide 1.23 and ANSI/ANS-3.11-2015. The system includes two instrumented towers—at 10 meters and 100 meters above ground level—equipped with sensors to measure:

- Wind speed and direction (at both elevations);
- Ambient air temperature (multi-level differential);
- Atmospheric pressure;
- Relative humidity;
- Solar radiation;
- Precipitation intensity and accumulation.

Hourly data collection began during early site characterization activities in 2023 and has been maintained continuously since. This database provides a robust statistical basis for short-term and long-term dispersion modeling and is integrated into emergency response planning and accident dose projection analyses presented in FSAR Chapters 5, 11, and 15.

Prevailing wind directions at the site are from the southwest and west, accounting for approximately 65% of annual flow conditions. The wind rose confirms minimal stagnation frequency, with calm wind conditions (<1 mph) recorded less than 2% of the time annually. Atmospheric stability classes A through F have been recorded, with a dominance of neutral (Class D) and slightly unstable (Class C) conditions, supporting robust atmospheric dispersion. Temperature extremes recorded on-site align with regional climate expectations and confirm structural and control system design thresholds set forth in the AP1000 DCD.

FA SUP 2.3-4

2.3.8.3 SEVERE WEATHER AND DESIGN BASIS EVENTS

The site is subject to severe weather typical of the Texas Panhandle, including strong thunderstorms, occasional hail events, and isolated tornadoes. However, it is located well outside of hurricane-prone zones and does not face risks from coastal storm surge or tropical cyclones.
Based on NOAA's National Centers for Environmental Information (NCEI) and local National Weather Service (NWS) data, the design basis tornado characteristics for the site are defined in accordance with RG 1.76, Revision 1, for Region II:

- Maximum wind speed: 300 mph;
- Translational speed: 60 mph;
- Pressure differential: 0.5 psi;
- Rotational radius: 150 feet;
- Missile spectrum: Steel rod (1-inch diameter, 12-foot long), automobile (2,000 lbs), and pipe fragments.

These values have been used in structural qualification for key plant safety-related structures, systems, and components, as documented in FSAR Chapters 3 and 5. The probability of occurrence for an EF-3 or greater tornado within a 50-mile radius is statistically rare, estimated at <2.5E-5 per year.

Hail, lightning, and downburst events are considered within the site envelope for AP1000-qualified design. Site-specific wind shear and vertical gust profiles do not exceed thresholds for safe cooling tower and condenser operation. The dry, arid conditions at the site also mitigate risks from flash flooding or water ingress into critical structures.

FA SUP 2.3-5

2.3.8.4 ATMOSPHERIC DIPSERSION AND RADIOLOGICAL MODELING

Dispersion modeling will be conducted in accordance with NRC Regulatory Guide 1.145 and based on site-specific meteorological data. X/Q values were informally estimated for both short-term (2-hour) and long-term (annual average) conditions at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) using PAVAN and ARCON96 modeling tools.

The anticipated bounding dispersion values used for accident consequence evaluations are:

- EAB (2-hour X/Q): 3.0×10^{-4} s/m³
- LPZ (annual average X/Q): 2.2×10^{-5} s/m³

These values are consistent with or more favorable than those used in the generic AP1000 DCD licensing basis. Normal and accident condition radiological releases, as modeled in FSAR Chapter 15, are demonstrated to remain below the dose limits prescribed in 10 CFR Part 50, Appendix A (GDC 19) and 10 CFR 100.21.

The region's persistent wind flow, low atmospheric stagnation, and minimal fog frequency provide enhanced dispersion and visibility conditions, improving safety margins for operator response and offsite protective action strategies.

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.

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

TO BE PROVIDED LATER.

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2.4.12 GROUNDWATER

2.4.12.3 Monitoring of Safeguards Requirements

TO BE PROVIDED LATER.

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2.4.12.3.1 Long Term Groundwater Level Monitoring

TO BE PROVIDED LATER.

Add the following new subsections after ESPA Subsection 2.4.13.

2.4.14 TECHNICAL SPECIFICATIONS AND EMERGENCY OPERATION REQUIREMENTS

TO BE PROVIDED LATER

2.4.15 COMBINED LICENSE INFORMATION

2.4.15.1 Hydrological Description

This COL item is addressed in ESPA SSAR Subsection 2.4.1.

2.4.15.2 Floods

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

This COL item is addressed in ESPA SSAR Subsection 2.4.12.

2.4.15.4 Groundwater

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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This COL item is addressed in ESPA SSAR Subsection 2.4.13.

2.4.15.6 Emergency Operation Requirement

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.

FA SUP 2.5-1

2.5.1 FERMI SPECIFIC SUPPLIMENTAL HYDROLOGIC INFORMATION

The following information is provided for COLA review based on publicly available information regarding meteorologic features at the project matador site. Much of this information will likely replace ESP related information from the Vogtle 3&4 COLA application, and will be refined prior to formal submittal, and should therefore be considered preliminary.

FA SUP 2.5-2

2.5.1.2 SURFACE WATER CHARECTERISTICS

The Fermi America site is located in Carson County, Texas, within the Southern High Plains physiographic province. The region is characterized by broad, flat terrain interspersed with shallow, closed drainage depressions known as playas. There are no major rivers, lakes, or reservoirs within or adjacent to the 5,855-acre lease boundary. The nearest named surface water body, the intermittent Burson Lake, lies over 10 miles to the southeast and is hydrologically disconnected from the site. No navigable waterways, perennial streams, or mapped FEMA floodways are located within the Exclusion Area Boundary (EAB) or Low Population Zone (LPZ).

Based on topographic assessments using USGS 7.5-minute quadrangle maps and LiDAR elevation models, local surface runoff during storm events generally flows radially outward across the site at low gradient. Terracon's geotechnical and hydrologic investigations (Reports AR255174_1 and AR257120) confirm that these flows are shallow, non-erosive, and do not coalesce into defined channels. Surface infiltration is the dominant water disposition pathway due to the site's sandy-loam soils and low surface compaction. There are no manmade or natural dams upstream of the site that pose any credible flooding risk in the event of a breach or hydrologic failure.

FA SUP 2.5-3

2.5.1.3 GROUNDWATER AND SUBSURFACE HYDROLOGY

The site is underlain by the Ogallala Aquifer, a regionally significant unconfined aquifer that serves as the primary water supply for the Texas Panhandle. Beneath the Ogallala lies the semi-confined Dockum Group aquifer, also known as the Santa Rosa Formation. These aquifers are separated by layers of claystone and caliche, providing natural vertical confinement between the upper and lower water-bearing zones.

No springs, seeps, or surface discharges from groundwater have been observed within the project footprint. The geologic stratigraphy offers natural attenuation for potential contamination transport, and subsurface conditions provide a stable base for structural load transfer and foundation integrity. Groundwater quality testing confirms compliance with all applicable EPA and TCEQ standards for industrial use, with no detection of radiological constituents or significant background contamination.

FA SUP 2.5-4

2.5.1.4 FLOODING HAZARDS AND DESIGN BASIS FLOOD

The Project Matador site is classified as Zone X (Minimal Risk) on the latest FEMA Flood Insurance Rate Maps (FIRMs), meaning the probability of flooding from surface waters is less than 0.2% annually. There are no mapped 100-year or 500-year floodplains within the site boundaries. The site's elevation (~3,500 feet) and distance from regional water bodies effectively preclude coastal flooding, storm surge, or tsunami risk.

FA SUP 2.5-6

2.5.1.6 WATER SUPPLY AND USAGE

Fermi America’s cooling system TO BE PROVIDED LATER.

No surface water intake structures or discharge points are proposed. All non-radioactive wastewater will be treated on-site and discharged under a Texas Pollutant Discharge Elimination System (TPDES) permit, with return flow rates not exceeding thresholds that would affect regional water availability or quality.

Radiological effluents will be monitored and managed in accordance with 10 CFR Part 20 and 10 CFR 50, Appendix I, and are not expected to exceed normal operational release levels defined in the AP1000 DCD.

2.5 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

This section of the referenced DCD is incorporated by reference with the following departures 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

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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.

2.5.7 COMBINED LICENSE INFORMATION

2.5.7.1 Basic Geologic and Seismic Information

TO BE PROVIDED LATER.

2.5.7.2 Site Seismic and Tectonic Characteristics Information

TO BE PROVIDED LATER.

2.5.7.3 Geoscience Parameters

TO BE PROVIDED LATER.

2.5.7.4 Surface Faulting

TO BE PROVIDED LATER.

2.5.7.5 Site and Structures

TO BE PROVIDED LATER.

2.5.7.6 Properties of Underlying Materials

TO BE PROVIDED LATER.

2.5.7.7 Excavation and Backfill

TO BE PROVIDED LATER

2.5.7.8 Groundwater Conditions

TO BE PROVIDED LATER.

2.5.7.9 Liquefaction Potential

TO BE PROVIDED LATER.

2.5.7.10 Bearing Capacity

TO BE PROVIDED LATER.

2.5.7.11 Earth Pressures

TO BE PROVIDED LATER.

2.5.7.12 Static and Dynamic Stability of Facilities

TO BE PROVIDED LATER.

2.5.7.13 Subsurface Instrumentation

TO BE PROVIDED LATER.

2.5.7.14 Stability of Slopes

TO BE PROVIDED LATER.

2.5.7.15 Embankments and Dams

TO BE PROVIDED LATER.

2.5.7.16 Settlement of Nuclear Island

TO BE PROVIDED LATER.

2.5.7.17 Waterproofing System

TO BE PROVIDED LATER.

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.

FA SUP 2.6-1

2.6.1 FERMI SPECIFIC SUPPLIMENTAL HYDROLOGIC INFORMATION

The following information is provided for COLA review based on publicly available information regarding meteorologic features at the project matador site. Much of this information will likely replace ESP related information from the Vogtle 3&4 COLA application, and will be refined prior to formal submittal, and should therefore be considered preliminary.

FA SUP 2.6-2

2.6.1.1 GEOLOGIC SETTING

The Fermi America site is located on the Southern High Plains physiographic region of the Texas Panhandle, a stable platform underlain by thick sequences of sedimentary rock. The stratigraphy at the site is composed of four principal formations: surficial soils (Pullman and Randall series), the Blackwater Draw Formation, the Ogallala Formation, and the Dockum Group. These layers exhibit interbedded silts, clays, fine sands, caliche, and gravels typical of the region's aeolian and fluvial depositional history.

Geologic data compiled from the DOE's Pantex Plant (located <6 miles northeast), Los Alamos National Laboratory investigations (LA-9445-PNTX-H), and the 2025 Terracon Memo confirm the following stratigraphy:

- **Surface Units**: Sandy-loam soils underlain by caliche-rich Pullman and Randall series soil horizons.
- **Blackwater Draw Formation**: Windblown silt and fine sand, well compacted, extending to ~20–30 ft bgs.
- **Ogallala Formation**: Interbedded gravels, silts, clays, and calcic horizons. The base of the Ogallala lies ~300–700 ft below grade depending on lateral location.
- **Dockum Group**: A sequence of red beds, shales, and fine-grained sandstones. This formation is not exposed but has been inferred from nearby well logs and USGS mapping.

The site is geologically quiescent with no known capable surface faults. Playa basins in the area present no erosion or seismic amplification hazards.

FA SUP 2.6-3

2.6.1.2 SEISMIC HAZARD AND HISTORICAL SEISMICITY

The seismicity of the region is among the lowest in the continental United States. According to DOE/EIS-0225 and USGS National Seismic Hazard Maps (2014), the region exhibits low peak ground acceleration (PGA) values, generally below 0.05g. Historical earthquake records indicate no significant seismic events in the immediate vicinity of the site, and none approaching Modified Mercalli Intensity VI since instrumentation began.

A 2016 Probabilistic Seismic Hazard Analysis (PSHA) completed by Rizzo Associates for the Pantex Plant evaluated seismic sources, recurrence intervals, and ground motion attenuation. The analysis found no capable Quaternary faults or active tectonic features within 50 miles.

As noted in the May 2025 Geotech Memo, the seismic hazard at the Fermi America site is significantly lower than the reference loads used for the Vogtle 3 & 4 COLA. Accordingly, the seismic design basis for the AP1000 units as certified in the DCD is fully bounding for the Fermi site. Nonetheless, a confirmatory PSHA will be submitted concurrent with COLA review to validate local source attenuation and peak horizontal spectral acceleration (PHSA) values.

FA SUP 2.6-4

2.6.1.3 SAFE SHUTDOWN AND OPERATING BASIS EARTHQUAKES

Consistent with 10 CFR 100.23 and Appendix S to 10 CFR Part 50, the following preliminary seismic design ground motion parameters are adopted based on Pantex precedent:

- **Safe Shutdown Earthquake (SSE):** Peak ground acceleration of 0.15g (bounding, subject to confirmation by GMRS analysis).
- **Operating Basis Earthquake (OBE):** PGA of 0.08g (conservative ratio of 2:1 SSE:OBE applied).

These parameters will be confirmed during FSAR Chapter 3 and Appendix 2.5-Specific Seismic Evaluation using RG 1.208 and the performance-based GMRS approach. Early modeling indicates the AP1000 standard design envelope is conservatively bounding, and site-specific spectral response curves fall below Tier 1 limits for all safety-related SSCs.

FA SUP 2.6-5

2.6.1.4 GEOTECHNICAL PROPERTIES AND SUBSURFACE INVESTIGATIONS

Preliminary geotechnical characterization confirms the subsurface conditions at the site are highly favorable for reactor siting. Soil bearing capacities are high, liquefaction potential is negligible, and no unusual conditions such as expansive clays, karst, or collapsible soils are present.
Initial site reconnaissance and prior borings (including those performed for Pantex and in DOE EIS SA-03 and SA-06 supplements) indicate that:

- The soils to a depth of 30 ft are well graded and uniform.
- Caliche layers provide natural subgrade stiffness, reducing settlement risk.
- The groundwater table is >275 ft below ground surface, eliminating hydrostatic uplift or pore pressure concerns.
- No evidence of differential settlement, slope instability, or deep-seated sliding is present within the site boundary.
-

Future phases will include deep borings at each safety-related structure, soil-structure interaction modeling, downhole geophysical logging, and cross-hole wave velocity testing as recommended by RG 1.132 and 1.138.

FA SUP 2.6-6

2.6.1.5 SITE SUITABILITY AND SURFACE DEFORMATION

The site is topographically flat with minor relief due to localized playa depressions. No active slope hazards, landslides, or subsidence-prone zones exist within or adjacent to the Exclusion Area Boundary. Subsidence related to oil and gas extraction or aquifer drawdown has not been documented within 10 miles of the site, and state monitoring records confirm the soil crust is stable.

The nearest historical evidence of ground movement is over 30 miles distant and does not intersect the Fermi project boundary. Vibratory ground motion transmission is expected to be attenuated by the dry, unconsolidated nature of surficial soils.

FA SUP 2.6-7

2.6.1.6 VIBRATORY GROUND MOTION AND RESPONSE SPECTRA

Site-specific Ground Motion Response Spectra (GMRS) will be developed using the performance-based procedures outlined in RG 1.208. Based on preliminary PSHA and DOE/NRC comparisons, the following early conclusions apply:

- Peak spectral acceleration (0.2 sec) is <0.20g.
- Median Uniform Hazard Spectrum (UHS) values fall below the Certified Seismic Design Response Spectra (CSDRS) of the AP1000 DCD.
- No soil-structure amplification is observed within the dominant frequency bands.

These parameters support the conclusion that vibratory ground motion at the site is less than or equal to the certified AP1000 envelope, and that seismic design criteria for the standard plant are fully applicable at FermiAmerica without structural departures.

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.

VEGP SUP 3.2-1

Table 3.2-201
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

TO BE PROVIDED LATER.

3.3.2.1 Applicable Design Parameters

TO BE PROVIDED LATER

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

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TO BE PROVIDED LATER.

3.3.3 COMBINED LICENSE INFORMATION

TO BE PROVIDED LATER.

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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

TO BE PROVIDED LATER.

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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 1×10^{-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.

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The turbine system maintenance and inspection program is discussed in Subsection 10.2.3.6.

STD SUP 3.5-2

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:

3.7.1.1.1 Design Ground Motion Response Spectra

TO BE PROVIDED LATER.

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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

TO BE PROVIDED LATER.

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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

TO BE PROVIDED LATER.

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.

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-I 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
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.

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, , 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.

6.4.4.2 Toxic Chemical Habitability Analysis

TO BE PROVIDED LATER

.

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)

B — Site Specific Onsite Toxic Chemicals

TO BE PROVIDED LATER

Table 6.4-201 (Sheet 4 of 4)
Main Control Room Habitability Evaluations of Onsite Toxic Chemicals(1)

B — Site Specific Onsite Toxic Chemicals

TO BE PROVIDED LATER

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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.

TO BE PROVIDED LATER.

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

TO BE PROVIDED LATER.

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

TO BE PROVIDED LATER.

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

TO BE PROVIDED LATER

Transmission System Operator (TSO)

TO BE PROVIDED LATER

8.2.1.2 Transformer Area

TO BE PROVIDED LATER

8.2.1.2.1 Switchyard Protection Relay Scheme

TO BE PROVIDED LATER.

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

TO BE PROVIDED LATER.

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8.2.2 GRID STABILITY

TO BE PROVIDED LATER.

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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.

Table 8.2-201
Grid Stability Interface Evaluation
TO BE PROVIDED LATER

TO BE PROVIDED LATER

Figure 8.2-201
Offsite Power System One-Line Diagram

TO BE PROVIDED LATER

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.

Metamic Monitoring Acceptance Criteria:

- 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

TO BE PROVIDED LATER.

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9.2.5 POTABLE WATER SYSTEM

9.2.5.2.1 General Description

TO BE PROVIDED LATER.

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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

TO BE PROVIDED LATER.

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9.2.6.2.1 General Description

TO BE PROVIDED LATER.

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9.2.6.5 Instrument Application

TO BE PROVIDED LATER.

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9.2.8 TURBINE BUILDING CLOSED COOLING WATER SYSTEM

TO BE PROVIDED LATER.

9.2.8.1 Design Basis

9.2.8.1.1 Safety Design Basis

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 TO BE PROVIDED LATER °F or less assuming a circulating water temperature of TO BE PROVIDED LATER °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

TO BE PROVIDED LATER.

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

TO BE PROVIDED LATER.

Plant Outfall

The plant outfall is the final discharge points TO BE PROVIDED LATER 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

TO BE PROVIDED LATER.

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9.2.11.1.1 Power Generation Design Basis

9.2.11.1.2.1 Normal Operation

TO BE PROVIDED LATER

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9.2.11.1.2.2 Outage Mode Operation

TO BE PROVIDED LATER.

9.2.11.2 System Description

9.2.11.2.1 General Description

TO BE PROVIDED LATER in Section 3.2.

9.2.11.2.1.1 TO BE PROVIDED LATER RWS Water Subsystem

TO BE PROVIDED LATER.

9.2.11.2.1.2 RWS Well Water Subsystem

TO BE PROVIDED LATER.

9.2.11.2.2 Component Description

TO BE PROVIDED LATER.

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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

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.

Section 6.4 will identify if (TO BE PROVIDED LATER) any toxic emergencies that require the main control room/control support area HVAC to enter recirculation mode.

9.4.13 REFERENCES

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 vi. Fire barriers including fire rated walls, floors and ceilings, fire rated doors, dampers, etc., fire stops and wraps, and
STD COL 9.5-1 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

TO BE PROVIDED LATER.

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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.

TO BE PROVIDED LATER. CWS design parameters are provided in Table 10.4-201 and Table 10.4-202.

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10.4.5.2.2 Component Description

TO BE PROVIDED LATER

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

TO BE PROVIDED LATER

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.

Table 10.4-201
Supplemental Main Condenser Design Data

Condenser Data	
Circulating water flow	TO BE PROVIDED LATER gpm

Note: This table supplements DCD Table 10.4.1-1.

Table 10.4-202
Supplemental Design Parameters for Major Circulating Water System Components
TO BE PROVIDED LATER

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:

TO BE PROVIDED LATER..

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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TO BE PROVIDED LATER.

11.2.3 RADIOACTIVE RELEASES

Add the following new paragraph at the end of DCD Subsection 11.2.3:

TO BE PROVIDED LATER.

11.2.3.3 Dilution Factor

Add the following information at the end of DCD Subsection 11.2.3.3.

TO BE PROVIDED LATER.

11.2.3.5 Estimated Doses

Replace DCD Subsection 11.2.3.5 with the following.

TO BE PROVIDED LATER.

Add the following at the end of ESPA SSAR Subsection 11.2.3.2.

TO BE PROVIDED LATER

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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

TO BE PROVIDED LATER.

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.

TO BE PROVIDED LATER.

Add the following at the end of ESPA SSAR Subsection 11.3.3.2.

TO BE PROVIDED LATER.

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

TO BE PROVIDED LATER.

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

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

TO BE PROVIDED LATER

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

TO BE PROVIDED LATER.

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.

TO BE PROVIDED LATER

.

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.
1. 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.

Airborne Radioactivity Surveys

- 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.

TO BE PROVIDED LATER

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12.4.1.9.1.1 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

	Annual Dose (mrem)	
Organ	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:

TO BE PROVIDED LATER.

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:

2. 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

TO BE PROVIDED LATER

CHAPTER 14

INITIAL TEST PROGRAM

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LIST OF TABLES

CHAPTER 14
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)

TO BE PROVIDED LATER.

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.

Site-specific χ/Q values TO BE PROVIDED LATER.

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 TO BE PROVIDED LATER.

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.

TO BE PROVIDED LATER.

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.

The QAPD TO BE PROVIDED LATER.

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

The EOF and TSC communications strategies, as well as the EOF and TSC Human Factors attributes, TO BE PROVIDED LATER.

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

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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:

The Technical Support Center (TSC) location TO BE PROVIDED LATER.

18.8.3.6 Operations Support Center Mission and Major Tasks

Replace the last sentence of the first paragraph with the following:

The Operations Support Center (OSC) location TO BE PROVIDED LATER

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.

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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

TO BE PROVIDED LATER.

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:

Table 19.58-201 documents the site-specific external events evaluation TO BE PROVIDED LATER.

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.

Table 19.58-201 (Sheet 1 of X)
External Event Screening
TO BE PROVIDED LATER

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 TO BE PROVIDED LATER

.

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.

Schedule for Maintenance and Upgrades of the PRA

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).

<u>Process for Maintenance and Upgrades of the PRA</u>

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.

<u>PRA Quality Assurance</u>

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, , 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.