North Anna 3 Combined License Application

Part 2: Final Safety Analysis Report

Revision 2
May 2009

REVISION SUMMARY

Revision 2

Section	Changes
Table 1.9-201	Revised to indicate conformance with SRP 11.4.II.10.
11.4.1	Incorporated a description of the long-term interim low-level radioactive waste storage space in the Radwaste Building and to identify the increased storage as a departure from the ESBWR DCD. Editorial change.
11.4.2.2.4	Revised to provide a description of, and requirements for, the long-term interim low-level radioactive waste storage space in the Radwaste Building, including an estimate of the amount of waste storage capacity, shielding for Class B and C waste storage, handling and integrity requirements, and requirements for crane design features.

Revision 1

Section	Changes
Chapter 1, 1.1-1-A, 1.8.2, 3.7.2.4, 3D, 3E, 6.1, 6.2.1.6, 8.2.4, 12.4.9, 13.6.2, 17.3	Updated titles and numbering to align with DCD R5.
1.1.1.6, 1.1.1.7, 1.1.1.11, 1.1.2.1, 1.1.2.2, 1.1.2.4, Table 1.1-201, 1.3, 1.6, Tables 1.6-201, 1.7-201, 1.7-202, 1.8-201, 1.8-202, & 1.8-203	Modified LMAs. Deleted NEI 03-12, Appendix F and NEI 06-06. Editorial changes added CDI entries for Zinc Injection System.
1.1.1.7, 1.1.1.9, 1.1.2.1, 1.1.2.2, 1.1.2.4, Table 1.1-201, 2.3-203, 2.5.4.10, 14.3A-1-1, 19.5, 19AA.2	Editorial updates/corrections.
1.1.1.7, Figure 9.5-201, 9A.1, 9A.3.1, 9A.4.7, Table 9A.5-7 Revisions, Table 9A.5-7 Departure	RAI NA3 09.05.01-17, Firewater Supply Locations
1.1.2.7	Revised estimated gross and net electrical power output.
1.1.2.8	Revised estimated key milestones.

Revision 1 *(continued)*

Section	Changes
Table 1.1-201, 1.8.3, 1.8.4, 1.8.201, 1.8.202, Tables 1.8-202 & 1.9-205, 1.10, 1.10-201, 1.10-202, Table 1.10-202, 2.0, 2.0-201, 2.0-203, Table 2.0-201, 2.1.2.1, 2.4.13, Section 2.5.1.2.3.k, Section 2.5.1.2.6.b, Section 2.5.1.2.6.g, Section 2.5.4.2.5.b Structural Fill, Section 2.5.4.5.2.b, 2.5.4.5.3, 2.5.4.8, Figure 2.5-253, 12.2-201, 12.2-202, 15.6	Revised to reflect issuance of ESP-003.
1.2.2.12.7, Table 1.8-203, 9.2.1.2	Added NAPS CDI for Plant Service Water System.
1.2.2.16.10	Updated action statement to align with DCD R5.
1.2.2.16.10, Tables 1.8-203, 1.10-201 & 3.2-1; Appendix 9A (Contents), 9A.1, 9A.3.1, 9A.4.7, 9A.5.12, 9A.7-2-A	Removed references to warehouse and cold machine shop (1.2.2.16.10). Added CDI for (no) cold machine shop (Table 3.2-1) and no warehouse, 9A1, 9A.2.1, 9A.3.1, 9A.4.7. Updated section number for Water Treatment Building (9a.5.12, Tables 1.8-203 & 1.10-201; 9A.7-2-A).
Table 1.9-202	Updated/corrected RGs 1.26 and 1.29.
Tables 1.9-202 & 1.9-203	RAI NA3 12.03-12.04-9, Editorial Corrections
1.3.1	Changed title of 1.3.1.
Tables 1.6-201, 1.9-201, & 1.9-203; 13BB	Updated NEI 06-13A to Rev. 1. Incorporated NEI 06-13A, Revision 1.
Table 1.6-201, 11.4.2.3.5, 11.4-201	Corrected NEI 07-10 title and revision.
Table 1.6-201, 12.2.2.4.2, Tables 12.2-15R, 12.2-18aR & 12.2-20aR	Deleted NEI 07-11 (Table 1.6-201). Editorial changes to align with RAI 11.02-1 response (12.2.2.4.2). Aligned with DCD R5 changes and added LMAs (Tables 12.2-15R, 12.2-18aR, & 12.1-20aR) RAI 11.02-1, Liquid Waste - Cost Benefit Analysis.
Table 1.6-201, 13AA.2.3, 13AA.2.4, 13BB	RAI NA3 13.02.01-1, NEI-06-13-A Revision 1 in FSAR
Table 1.6-201, 17.5, 17.5-202	Specified QAPD tie to NEI 06-14A.
Table 1.6-201, 17.6.3	RAI NA3 17.06-1, Maintenance Rule
Tables 1.8-201, 12.2-18bR & 12.2-203	RAI NA3 12.02-10, Clarification of FSAR Tables in Chapter 12, FSAR Table 12.2-17R Update w/Data on Radionuclide Ratios

Revision 1 *(continued)*

Section	Changes
Tables 1.8-202 & 1.10-201, 2.0, 2.0.1, Tables 2.0-2R & 2.0-201, 2.3.5.1, Tables 2.3-208 thru 2.3-215, 2A, Table 2A-4R	Updated to align with DCD R5.
Tables 1.8-202 & 2.0-201	RAI NA3 15.06.05-1, Radiological Consequence Doses - Evaluation Factors
Table 1.8-202; 12.2.2.2.2, 12.2.2.2.6, 12.2.2.4.2, 12.2.2.4.4; Tables 12.2-15R, 12.2-17R, 12.2-18bR, 12.2-201, 12.2-203, & 12.2-204	RAI NA3 12.02-1, Dose Analysis
Tables 1.8-203 and 1.10-201, 11.2, 11.2.2.3, 11.4, 11.4.2.3.5, 11.4-1-A, Table 11.5-201	Changed “mobile” liquid and solid radwaste systems to “process” systems.
1.9.2, 1.9.3, Tables 1.9-201, 1.9-202, 1.9-203, 1.9-204, 1.9-205, and 1.10-202, 1.11.1, 1C.1	Miscellaneous clarifications and corrections.
Table 1.9-201	Updated evaluation for SRP Section 6.5.1 to conform to DCD R5 changes. RAI NA3 08.02-18, GDC-2 Applicability, RAI NA3 08.02-20, BTP 8-3 Applicability, RAI NA3 08.02-21, BTP 8-5 Applicability, RAI NA3 08.02-22, BTP 8-6 Applicability, & RAI NA3 17.05-1, Comparison of QAPD and SRP 17.5 Criteria.
	Revised evaluation of BTP 8-2 to align with DCD R5.
Tables 1.9-201 and 1.9-202	Revised conformance evaluation for SRP 5.4.13 acceptance criterion 4 (Table 1.9-201) and for RG 1.93 (Table 1.9-202).
Tables 1.9-201, 1.9-203 & 1.10-201	Updated references to DCD R5. Editorial corrections.
Table 1.9-201	Updated turbine model number.
Tables 1.9-201, 1.9-202, & 1.9-204, 14.2.9.1.3	RAI NA3 14.02-5, Personnel Monitors and Radiation Survey Instruments
Tables 1.9-201 & 1.9-202	RAI NA3 14.02-6, Site-Specific Preoperational Test
Table 1.9-201, 13.1.1.2.1, 14AA.2.2.10, 17.5, 17AA	QA Policy incorporated in QAPD.

Revision 1 *(continued)*

Section	Changes
Table 1.9-202	Changed RG 1.29 commitment from Rev. 4 to Rev. 3. Changed RG 4.15 commitment from Rev. 2 to Rev. 1. Editorial changes.
	Changed RG 1.40 to "Conforms" and RG 1.136 to reflect DCD R5 corrections.
	RAI NA3 03.02.01-3, RG 1.29 Revision Clarification
	RAI NA3 08.03.02-2, RGs 1.41, 1.128, 1.129 Conformance Clarification
Tables 1.9-202 & 1.9-204	Added an exception to RG 1.8 in Table 1.9-202; revised NQA-1 year/title in Table 1.9-204.
Table 1.9-202, 3.9.2.4	RAI NA3 03.09.02-2, FIV Program Schedule for Reactor Internals
Table 1.9-202. 13.1.1.2.1, 13.1.1.2.10, 13.1.2.1, 13.1.2.1.1, 13.1.2.1.1.2, 13.1.2.1.1.9, 13.1.2.1.1.10, 13.1.2.1.5, Table 13.1-201, Figure 13.1-204	RAI NA3 13.01.02-13.01.03-1, Fire Protection Organization
Table 1.9-202, 17AA	RAI NA3 03.02.02-1, RG 1.26 Revision Clarification
Table 1.9-203	Added conformance evaluations for RG Positions C.III.1.5.4.3 through C.III.1.5.4.13.
Table 1.9-203	RAI NA3 14.03.10-1.4, ITAAC for Offsite Full Participation Exercise
Table 1.9-204	RAI NA3 09.05.01-9, COLA Reference to NFPA 55
	Added NERC standards.
Table 1.9-204, 2.3.1.3.1, 2.3-204, 2.3-205, 2.3-206	RAI NA3 02.03.01-1, Wind Speed Values
Table 1.9-204, 2.3.2.3.1, 2.3.2.3.2, Section 2.3 References	RAI NA3 02.03.02-1, Local Meteorology
Table 1.9-205, 2.2.3.1.1, 2.2-213, 2.2-214, 2.2-215	RAI NA3 02.02.03-1, Explosion Hazard - Underground Gasoline Storage Tanks
Table 1.10-201	Updated to align with DCD R5 changes; revised COL Item 12.3-3-A from applicant to holder.
	Corrected referenced section for COL Item 8.2.4-5-A.
Table 1.10-201, 3.6	Deleted COL Item 3.6.5-1-A
Table 1.10-201, 3.11.4.4, 3.11.7, 3.11-1-A	Added reference to DCD EQ Program description. Administrative changes to reflect DCD R5 numbering and title changes.

Revision 1 *(continued)*

Section	Changes
Table 1.10-201, 4.3.3.1, 4.3-1-A, 4A.1	Editorial changes to align with DCD R5; revised COL items 4.3-1-A and 4A-1-A.
Table 1.10-201, 5.2.4, 5.2.4.11, 5.2.5, 5.2-1-A, 5.2-2-H, 5.2-3-A	Revised 5.2-1-H to 5.2-1-A. Added Section 5.2.5 to COL Item 5.2-2-H. Added COL Item 5.2-3-A and updated associated content accordingly. Updated to align with DCD R5.
Table 1.10-201, 5.2.4.3.4, 5.2.4.6, 5.2-1-A, 6.6.6	Editorial corrections related to COL Item 5.2-1-A.
Table 1.10-201, 5.3.1.5	Revised for future submittal of PTLR curves.
Table 1.10-201, 6.1	Incorporated deletion of COL Item 6.1.3-1-A in DCD R5.
Table 1.10-201, 6.2.4.2, 6.2-1-H	Updated to align with DCD R5 changes related to COL Item 6.2-1-H.
Table 1.10-201, 6.6, 6.6.2, 6.6.7, 6.6.7.1.1, 6.6.7.1.2, 6.6.7.1.4, 6.6.7.1.5, 6.6.7.1.6, 6.6.7.1.7, 6.6-2-A, 6.6.12	RAIs NA3 10.03.06-1, FAC - Construction Phase, 10.03.06-2, FAC - Baseline Thickness, and 14.02-1, Initial Plant Test - Switchyard Components. Added COL Item 6.6-2-A to align with DCD R5. Added weld accessibility controls description.
Table 1.10-201, 9.1.1.7, 9.1.4.13, 9.1.4.19, 9.1.5.8, 9.1-4-A	Added Section 9.1.1.7. Revised COL Item 9.1.6-4-A to 9.1-4-A to align with DCD R5.
Table 1.10-201, 9.2.5, 9.2.5-1-H	COL Item 9.2.5-1-A changed to 9.2.5-1-H in DCD R5.
Table 1.10-201, 9.5.1.12, 9.5.1.15.3, 13.1-1-A, Appendix 13AA	Editorial changes to align with DCD R5 related to deleting STD SUP 9.5.1-2 and adding COL Items 9.5.1-7-H and 13.1-1-A.
Table 1.10-201, 9.5.1.15.2, 9.5.1-9-A	RAI NA3 09.05.01-1, Fire Protection Program Change Process
Table 1.10-201, 9.5.2.2, 9.5.2.5-1-A, 9.5.2.5-2-A, 9.5.2.5-3-A, 9.5.2.5-4-A, 9.5.2.5-5-A	Changed COL Item 9.5.2.5-1-A to 9.5.2.5-3-A. Added COL Items 9.5.2.5-4-A and 9.5.2.5-5-A.
Table 1.10-201, 10.2.3.4, 10.2.5	Added description of plant-specific turbine maintenance and inspection program. Acknowledged permission to use bounding property values in turbine missile evaluations until actual material specimens are available.
Table 1.10-201, 11.4.1, 11.4.2.3.5, 11.4-1-A, 11.4-2-A, 11.4-3-A	Updated to align with DCD R5. Editorial corrections.
Table 1.10-201, 11.5.7	Deleted references to Section 12.2.
Table 1.10-201, 11.5.4.6, 11.5.4.7, 11.5-1-A, DCD Table 11.5-2, DCD Table 11.5-4	Editorial corrections related to title changes and to add a description of process radiation monitoring procedures.
Table 1.10-201, 12.2.1.5, 12.2-4-A	RAI NA3 12.02-4, STD SUP 12.3-4-A Not Included
Table 1.10-201, 12.5-2-A	Changed title of COL Item 12.5-2-A.

Revision 1 *(continued)*

Section	Changes
Table 1.10-201, 12BB, 13.6.5, 16.0.1, 16.0-1-A, 16.0-2-H	Editorial corrections. Updated to align with DCD R5 COL Items 16-0-1-A & H, and to address NEI template 07-03 in Appendix 12BB.
Table 1.10-201, 13.6.1.1.3, 13.6.1.1.5, 13.6.1.1.8, 13.6.2, 13.6.3	Updated to align with DCD R5 changes. Added 10 new COL items to Section 13.6.
Tables 1.10-201 & 13.4-201, 6.6, 6.6.2, 6.6.7.1	Added new COL Item. RAI NA3 10.03.06-1, FAC - Construction Phase (Added description of augmented ISI program). RAI NA3 10.03.06-2, FAC - Baseline Thickness (Added discussion of controls to ensure accessibility for PSI and ISI NDE. Added reference to FAC program.)
Table 1.10-201, 14.2.2.1, 14.2.2.2, 14.2.7, 14.2.9, 14.2.10	Updated to align with DCD R5 changes related to new COL Items 14.2-1-1 and 14.2-5-A.
Table 1.10-201, 14.3A	Added Appendix 14.3A to align with DCD R5.
Table 1.10-201, 17.4.1, 17.4.6, 17.4.9, 17.4.10, 17.4-1-H	Updated to reflect DCD R5 changes to COL Item 17.4-1-A.
Table 1.10-201, 18.13, 18.13.3, 18.13.5	Added COL Item 18.13-1-H.
2.0, Tables 2.0-2R, 2.0-201 thru 2.0-203, Figures 2.0-201 thru 2.0-207, 2.1.1.1, 2.1.1.2, 2.1.2.1, Figure 2.1-201, 2.2.2.6.1, 2.2.2.6.2, 2.2.3, 2.2.3.2.2, 2.2.3.4, Tables 2.2-201 thru 2.3-204, Figure 2.2-201, 2.3.1, 2.3.1.3.4, 2.3.2, 2.3.2.3, 2.3.3, 2.3.3.1.2, 2.3.4.1, 2.3.4.3, 2.3.5, 2.3.5.1, Tables 2.3-17R thru 2.3-203, 2.3-201	Editorial corrections.
Table 2.0-201	RAI NA3 02.03.01-3, Clarification of Ambient Temperatures
	RAI NA3 02.05.04-6, Allowable Dynamic Bearing Capacity Differences
Table 2.0-201, 2.3.3.1.2, 2.3.4.1	Updated tallest structure information.
Table 2.0-201, Figure 2.3-201	Updated to reflect GEH analysis. Added Fuel Building information, added Radwaste Building unfiltered inleakage information, deleted Fuel Building Cask Doors information, and added Reactor Building TSC information.
Table 2.0-201, 2.3.1.2, 2.3-207	RAI NAPS 02.03.01-2, 10 CFR 52.79(a)(1)(iii) Dry/Wet Bulb Temperatures

Revision 1 *(continued)*

Section	Changes
Tables 2.0-201, 2.3-15R, 12.2-18bR, 12.2-201 & 12.2-203, 2.3.5.1, 12.2.2.4.4	RAI NAPS 02.03.05-2, Clarification of χ/Q and D/Q Values, FSAR Table 2.3-16R vs. ER Table 2.7-4 re: EQ
Figure 2.0-205	Updated building coordinates to align with DCD R5.
Figure 2.1-201	Updated to align with DCD R5 (cooling tower pond, construction zones, and plot plan background).
Table 2.2-202	Added Ancillary Diesel Building data.
Tables 2.2-202, 2.2-203, & 2.2-204	Updated chemicals and chemical quantities for Unit 3 and removed Units 1 & 2 chemicals.
2.3.2.3.2	Clarification of RAI NA3 02.03.02-1, Local Meteorology, response.
2.3.4.3	Added TSC and renumbered Table 2.3-205 to 2.3-207.
2.3.5.1	RAI NA3 02.03.05-1, χ/Q and D/Q Values
2.3.5.1, Table 2.3-15R	Updated receptor distances.
2.3.5.1, Tables 2.3-204 thru 2.3-215	RAI NA3 02.03.05-3, Long Term (Routine) Diffusion Estimates
Tables 2.3-201 thru Tables 2.3-207	Updated to reflect GEH analysis. Inserted two new tables.
2.4.1, 2.4.1.1, 2.4.2, 2.4.2.2, 2.4.2.3, 2.4.3, 2.4.4, 2.4.5, 2.4.6, 2.4.7, 2.4.7.2, 2.4.7.4, 2.4.7.5, 2.4.7.6, 2.4.8, 2.4.9, 2.4.10, 2.4.11, 2.4.11.5, 2.4.11.6, 2.4.12, 2.4.12.1.2, 2.4.12.1.3, 2.4.12.3, 2.4.12.4, 2.4.13, 2.4.14, Tables 2.4-15R thru 2.4-17R, Tables 2.4-201 thru 2.4-212, 2.5.1, 2.5.1.2.3, 2.5.1.2.6, 2.5.1.2.7, 2.5.2, 2.5.2.5, 2.5.2.6.7, 2.5.2.6.8, 2.5.2.6.9, 2.5.2.6.10, 2.5.4, 2.5.4.3, 2.5.4.5.3, 2.5.4.5, 2.5.4.6, 2.5.4.6.3, 2.5.4.7, 2.5.4.7.1, 2.5.4.7.2, 2.5.4.7.4, 2.5.4.8, 2.5.4.10, 2.5.4.10.1, 2.5.4.10.2, 2.5.4.11, 2.5.4.12, 2.5.5, 2.5.5.1.2, 2.5.5.1.3, 2.5.5.2.3, 2.5.5.2.4, 2.5.5.3, 2.5.6, Tables 2.5-201 thru 2.5-219, Figures 2.5-201 thru 2.5-276	Miscellaneous editorial changes (LMAs, delimiters).
2.4.2.3, Tables 2.4-201 thru 2.4-204, Figures 2.4-201, 2.4-203, 2.4-204, & 2.4-206 thru 2.4-216	Updated to align with DCD R5; revised Section 2.4 based on DCD R5 impacts.
2.4.14	Corrected typographical error.

Revision 1 *(continued)*

Section	Changes
Tables 2.4-15R	Added note explaining WP-3 “?” value.
2.5.4.5.3	RAI NA3 02.05.04-3, Material and Engineering Properties of Backfill
2.5.4.8, 2.5.4.10, Table 2.5-213	Corrected seismic classification of Turbine Building to align with DCD R5.
2.5.4.10, Tables 2.5-213 & 2.5-215, Figures 2.5-209 thru 2.5-215, 2.5-221, 2.5-222, 2.5-229 thru 2.5-234, 2.5-252, 2.5-255	Updated to align with DCD R5.
Table 2.5-213	Updated Radwaste Building seismic reference.
2.5.4AAS1, 2.5.4AAS2	Revised title on link page. Added MACTEC Geotechnical Data Report Supplement 2.
3.2, 4.2, 9.3.10.2, 9.5.1.4	Added metric values and deleted STD COL 4.2.6 from Section 4.2.
3.7.1.1, 3.12	Editorial changes.
3.7.2.8	Updated action statement to account for DCD R5 changes.
3.9.3.7.1(3)e, 3.9.3.7.1(3)f, 3.9.6, 3.9.6.1, 3.9.6.1.4, 3.9.6.1.5, 3.9.6.5, 3.9.6.6, 3.9.6.7, 3.9.6.8, 3.9.8, 3.9.10, Table 13.4-201	Expanded IST Program Description.
3.9.3.7.1(3)f, 3.10.1.4, 3.11.2.2, 3.11-1-A, 3.12	Added supplement separator line. Corrected EQD definition. Added dotted lines to signify supplement information within a supplement.
3.9.3.7.1(3)f	RAI NA3 03.09.03-2, Update Reference to Snubber ITAAC Table
3.9.6.8	RAI NA3 03.09.06-3, Dynamic/Static Testing of AOVs
	Clarified IST description for other than air-operated, power-operated valves.
3.10.1.4, 3.10.4	Added commitment to provide an implementation schedule for seismic and dynamic qualification of mechanical and electrical equipment. Updated title to DCD R5.
3.11-1-A	Editorial correction.
3.11.4.4	RAI NA3 03.11-1, EQ Process Implementation; RAI NA3 03.11-2, DCD EQ Approach Implementation; & RAI NA3 03.11-3, Additional EQ Approach Implementation
4.2, 4.3, 4A	Editorial changes.
4.2	Revised to be all IBR. Editorial changes. Deleted COL Item 4.2.6.

Revision 1 *(continued)*

Section	Changes
5.2.1.1	RAI NA3 05.02.01.01-1, ASME BPV Code + ASME Code for O&M
5.2.1.2	RAI NA3 05.02.01.02-1, Code Cases Not in EWBWR DCD re: ASME BPV or OM Codes
5.2.4, 5.2.4.2	RAI NA3 05.02.04-3, PSI Exams Equivalent to Inservice Inspection (ISI) Exams
5.2.4.3.4, 5.2.4.6, 6.6.6	RAI NA3 05.02.04-4, Incorporating Limits of 10 CFR 50.55a(b)(2)
5.2.5.9	RAI NA3 05.02.05-1,Leak Detection Monitoring
	Restored sentence proposed to be deleted per RAI 05.02.05-1.
5.3.1.5	Added 5.3.1.5 to include a commitment to PT LR.
5.3.1.8, 5.3.1.8.1, 5.3.1.8.2, 5.3.1.8.3, 5.3.1.8.4, Table 5.3-201	Revised 5.3.1.8 and added Table 5.3-201 to include information provided in response RAI NA3 05.03.01-1, Reactor Vessel Surveillance Capsule Program.
6.2.4.2, 6.4.4	Corrected LMA. Editorial change.
6.4.5	Revised action statement to delete last paragraph of DCD Section 6.4.5.
	Updated to reflect GEH analysis.
6.6.7.1.3	Replaced “initial inspections” with “preservice inspections.”
6.6.10.2	Editorial changes.
6B	Updated title per DCD R5.
6D	Added Appendix 6D.
6E, 6G, & 6I	Added appendices 6E, 6G, & 6I.
6F	Added Appendix 6F.
6H	Added to reflect DCD R5 addition of Appendix 6H.
8.2.1.2	RAI NA3 08.02-2, Cable Routing Intermediate Switchyard; & NA3 RAI 08.02-4, Potential Cable Degradation
	RAI NA3 08.02-29, Underground Cable Testing
8.2.1.2, 8.2.1.2.1, 8.2.1.2.2, 8.2.2.1, 8.2.3, 8.2.4-5-A, 8.2-201, 8.2-202, Figures 8.2-202 & 8.2-203, 8.3.2.1.1, 8A.2.1	Editorial corrections. Added 8.2.3.
8.2.1.2.1	RAI NA3 08.02-25, Surge and Lightning Protection Description
8.2.1.2.2	RAI NA3 08.02-7, Protective Relay Acceptance

Revision 1 *(continued)*

Section	Changes
8.2.1.2.3	RAI NA3 08.02-8, Industry Standards for Switchyard; & NA3 RAI 08.02-9, Transformer Testing Inclusion
8.2.2.1	RAI NA3 08.02-13, Clarify Tech Spec Reference
	RAI NA3 08.02-32, 34.5 kV Loads Impact on Grid Stability
Figure 8.2-201	RAI NA3 08.02-1, Switchyard Figure Discrepancy
	RAI NA3 08.02-30, Identify Switchyard Transformers
Figures 8.2-201 & 8.2-202	Added new bay to connect 500 kV Ladysmith line.
8.3.2.1.1, 8.3.5, 8.3-201	RAI NA3 08.03.02-1, SBO Response Procedures
9.1.4.13, 9.1.4.19	Editorial changes.
9.1.5.6	RAI NA3 09.01.05-1, Size and Rating Requirements for Slings
9.1.5.9, 9.1-5-A	RAI NA3 09.01.05-2, Heavy Load Equipment Outside Scope of DCD
9.2.1.2, 9.2.4.2, 9.2.4.3, 9.2.4.5, Figure 9.2-203, 10.4.5.2.3, Table 11.5-201	RAI NA3 11.05-2, Process and Effluent Monitoring
9.2.1.2; Tables 9.2-201, 9.2-202, 9.2-203, & 9.2-204; Figures 9.2-201, 9.2-202, 9.2-203, 9.2-204, & 9.2-205; 9.3.9.1, 9.3.9.2, 9.3.9.2.1, 9.3.9-2-A, 9.5.1.4, 9.5.1-1-A, DCD Table 9.5-2, 9.5.4.2, 9A.4.7	Corrected and added LMAs. Corrected section titles. Added commitment to update FSAR with detailed fire hazards analysis information.
9.2.1.2	RAI NA3 09.02.01-3, PSWS Material Selections Based on Water Quality
9.2.1.2, Table 9.2-201	Updated to align with DCD R5 related to valve and strainer terminology, cooling tower capacity, and elimination of AOVs.
9.2.3.2	Aligned terminology with DCD R5 related to shutdown/refueling/ startup and water storage tanks.
Figure 9.2-201	RAI NA3 09.02.01-1, Cooling Tower Performance Capability
Figures 9.2-202 & 9.2-203	Deleted the Potable Water System connection to the Turbine Building. Added a PWS connection to the Ancillary Diesel Building. Changed Security Building to Guard House, Intake Structure to Station Water Intake Building, and Hot/Cold Machine Shop to Hot Machine Shop (Figure 9.2-202). Changed Security Building to Guard House, Hot/Cold Machine Shop to Hot Machine Shop, and deleted the Sanitary Waste Discharge System connection to the Turbine Building (Figure 9.2-203).

Revision 1 *(continued)*

Section	Changes
Figure 9.2-204	Revised to reflect Plant Cooling Tower Makeup System design changes.
9.3.2.2	RAI NA3 09.03.02-1, Sampling Containment Atmosphere
9.5.1.4	RAI NA3 09.05.01-8, Quality of Fire Water Sources
9.5.1.4, Figures 9.5-202 and 9.5-203	Updated to align with DCD R5 changes related to the capacity of the secondary firewater source. Added LMAs.
9.5.4.2	RAI NA3 09.05.01-15, Fire Barrier Testing
	Editorial changes.
Table 9.5-201	Added NFPA codes and NEIL.
Figure 9.5-201	Deleted Cold Machine Shop & Office Building, and updated general arrangement.
Figure 9.5-202	Changed “Intake Structure” to “Station Water Intake Building” and updated general arrangement.
Figure 9.5-203	Added Cooling Tower Maintenance Building, Hybrid Cooling Tower Electrical Building, and Dry Cooling Tower Electrical Building.
9.5.1.15.6	RAI NA3 09.05.01-5, Control of Combustibles in Rooms Adjacent to MCR; RAI NA3 09.05.01-6, Control of Combustibles Below Floor in MCR Complex; RAI NA3 09.05.01-7, Control of Combustibles in Computer Rooms; & RAI NA3 09.05.01-13, Storage of Hazardous Chemicals
9.5.1.15.6, 9.5.1-8-A	Aligned titles with DCD R5.
9.5.1.15.9	RAI NA3 09.05.01-11, Fire Protection Program QA
9.5.4.2	Added treatment of Ancillary Diesel Generators.
	RAI NA3 09.05.04-2, Diesel Fuel Oil for Seven-Day Loaded Run
	RAI NA3 09.05.04-4, Fuel Oil Transfer System Corrosion Control
	Updated to align with DCD R5 related to material and corrosion protection for underground systems; and editorial changes to RAI NA3 09.05.04-4 markups.
	RAI NA3 09.05.04-6, Corrosion Protection Systems
9.5.5	Corrected title to agree with DCD.
9A.1, 9A.3.1	Deleted reference to Station Water Pump House.
9A.2.1	Deleted reference to Tables 1.9-202 and 1.9-203.

Revision 1 *(continued)*

Section	Changes
Table 9A.5-7 Revisions	Revised applicable fire areas.
	Added F7500 to deleted fire area list. Removed Table 9A.5-7 Departure added by RAI NA3 09.05.01-17, Fire Water Supply Locations.
Table 9A.5-7R	Completed to-be-done items with available information and updated design basis fire impact on safe shutdown. Added Fire Areas F7155, 7165, 8182 & 8201.
Figure 9A.2-33R	Revised site plot plan.
Figures 9A.2-201 thru 9A.2-204	Updated general arrangement; added LMA.
Figures 9A.2-205 & 9A.2-206	Deleted "Cold" machine shop; updated general arrangement; added LMA.
9A.5.12	Clarified commitment item.
10.2.3.4	Updated turbine model number.
10.2.3.6	Section inserted (new COL Item 10.2-1-A, Turbine Rotor Maintenance).
10.2.3.8	Section inserted (new COL Items 10.2-2-A, Turbine Missiles.
10.4.5.2.1, 10.4.5.2.2	RAI NA3 10.04.05-1: Circulating Water Large Bore Piping Codes and Failures
10.4.5.5	RAI NA3 10.04.05-2: Flooding due to Hybrid Cooling Tower Failure
	Corrected CW minimum inlet temperature.
10.4.5.6	Inserted Section title.
Table 10.4-3R	Changed to reflect DCD R5 revisions.
Table 10.4-201	Corrected units of conductivity.
Figures 10.4-201, 10.4-202, & 10.4-203	Added LMAs. Editorial changes deleted reference to NEI Topical Reports not incorporated by reference.
11.2.1	RAI NA3 11.02-1, Liquid Waste - Cost Benefit Analysis
	RAI NA3 11.03-2, Cost Benefit for GWMS
11.2.2.3.3	Changed action statements to agree with DCD R5 modifications.
	RAI NA3 11.02-2, LWMS: Sampling Non-Radioactive Systems
11.3.1	RAI NA3 11.03-0, Gaseous Waste - Cost Benefit Analysis
11.4.1	RAI NA3 11.04-1A, Solid Waste - Cost Benefit Analysis

Revision 1 *(continued)*

Section	Changes
11.4.2.3.5	RAI NA3 11.04-2, SWMS: Sampling Non-Radioactive Systems
11.5.4.9	Added "sampling and analytical" to "frequencies" with respect to discussion radioactive gaseous and liquid wastes.
Table 11.5-201	Revised Note 1
12.1.1.3.1, 12.1.1.3.2, 12.1.1.3.3, 12.1.3, 12.1-1-A, 12.1-2-A, 12.1-3-A, 12.1-4-A	Added supplements to address ALARA DCD COL Items 12.1-4-A, 12.1-1-A, 12.1-2-A, & 12.1-3-A.
12.2.1.5	RAI NA3 12.02-6, Additional Contained Source Uses
	Corrected LMA delimiters to reflect Section 12.2.1.5, other Contained Sources, as DCD item.
12.2.2.4.4	Updated distance from ISFSI to nearest residence.
12.2.2.4.4, Table 12.2-203	RAI NA3 12.02-2, Dose Analysis and EPA Standards
	Changed ISFSI number of casks and dose contribution, and changed existing units and site total doses.
	RAI NA3 12.02-12, Dose Contributions
Table 12.2-18bR	Editorial clarifications to Note 4.
12.3, Tables 12.2-20bR & 12.2-201, 12A	Deleted LMA. Corrected table values from mSv to mrem. Corrected dose rate units. Editorial changes.
12.4.7.1	Changed section number to align with DCD Section 12.4 R5 changes.
12.5, 12.5.4	Editorial changes.
Tables 12.2-15R, 12.2-18bR & 12.2-204	RAI NA3 12.02-11, Clarify Information In Section 12 Tables
Tables 12.2-17R & 12.2-19bR	RAI NA3 12.02-3, Liquid Dose Offsite
12B	Added to reflect DCD R5 addition of Appendix 12B.
12BB	RAI NA3 12.03-12.04-2, Very High Radiation Area Drawings; and RAI NA3 12.05-2, Site-Specific Alterations to NEI 07-03
	Editorial
13.1, 13.1.1, 13.1.2.1.1.9, 13.1.2.1.1.12, 13.1.2.1.5, 13.1.3.1, Table 13.1-201, Figure 13.1-201, 13.6.2, 13AA, 13AA.1.4, 13AA.2.3	Corrected LMAs. Updated executive titles. Revised to specifically address NAPS ESP COL 13.6-1.
13.1.1	RAI NA3 17.05-7, Making Changes to Organizational Descriptions

Revision 1 *(continued)*

Section	Changes
13.1.1, 13.1.1.1, 13.1.1.2	RAI NA3 13.01.01-3, Corporate Organization
13.1.1, 13.1.1.2.10, 13.1.1.3.1.5, Figures 13.1-201 & 13.1-205	Updated corporate structure and responsibilities.
13.1.1.2, 13.1.1.2.1, 13.1.1.2.9, 13.1.1.3.1.7, 13.1.1.3.1.8, 13.1.1.3.2, 13.1.1.3.2.1, 13.1.1.3.2.2.1, 13.1.1.3.2.2.2, 13.1.1.3.2.2.3, 13.1.1.3.2.2.5, 13.1.2.1.1, 13.1.2.1.1.1, 13.1.2.1.1.2, 13.1.2.1.1.3, 13.1.2.1.1.8, 13.1.2.1.2, 13.1.2.1.2.1, 13.1.2.1.2.2, 13.1.2.1.2.3, 13.1.2.1.2.6, 13AA.1.9	Added component and project engineering. Changed SNSOC to FSRC. Revised the corporate director of nuclear engineering position description. Corrected the reporting relationship for the EPC in Appendix 13AA. Corrected/updated the reporting relationships in Figures 13.1-203 and 204. Resequenced the operations department functions (13.1.2.1.2).
13.1.1.2.1	RAI NA3 13.01.01-1, Guidance Regarding Outside Company Work
13.1.1.2.10	RAI NA3 13.02.02-1, SRP Section 12.2.2 re: Section 13.1
13.1.1.3, 13.1.1.3.1, 13.1.1.3.1.1, 13.1.1.3.1.2, 13.1.1.3.1.3, 13.1.1.3.1.4, 13.1.1.3.1.6, 13.1.1.3.1.7, 13.1.1.3.1.8, 13.1.1.3.2, 13.1.1.3.2.1, 13.1.1.3.2.2, 13.1.1.3.2.2.1, 13.1.1.3.2.2.2, 13.1.1.3.2.2.3, 13.1.1.3.2.2.4, 13.1.1.3.2.3, 13.1.1.3.2.4, 13.1.1.3.2.5, 13.1.1.3.2.6, 13.1.1.3.2.7, 13.1.1.3.2.8, 13.1.1.3.2.9	RAI NA3 13.01.01-2, Executive and Management Positions
13.1.2.1	RAI NA3 13.01.02-13.01.03-3, Plant Organization regarding Section 17.5
13.1.2.1.1.3	RAI NA3 13.01.01-4, Plant Maintenance Programs
13.1.2.1.2.2, 13.1.2.1.2.3	RAI NA3 13.05.02.01-2, Procedures in FSAR Section 13.5.2
13.1.2.1.5	RAI NA3 09.05.01-12, Fire Brigade Leader Qualifications
Figure 13.1-204	RAI NA3 13.01.01-6, Organizational Arrangement Regarding Nuclear w/ Corporate
13.3	Updated to align with DCD R5.

Revision 1 *(continued)*

Section	Changes
13.5, 13.5.1, 13.5.2, 13.5.2.1, 13.5.2.1.1, 13.5.2.1.2, 13.5.2.1.3, 13.5.2.1.4, 13.5.2.1.5, 13.5.2.1.6, 13.5.2.1.7, 13.5.2.2.1, 13.5.2.2.2, 13.5.2.2.3,13.5.2.2.4, 13.5.2.2.5, 13.5.2.2.6, 13.5.2.2.6.2, 13.5.2.2.6.4 13.5.2.2.6.5, 13.5.2.2.7, 13.5.2.2.8, 13.5.2.2.9, 13.5-5-A, 13.5-5-A, 13.5-6-H	Corrected LMA applicability and delimiter notations. Revised 13.5.2.2.6.5 to reference Section 9.1.5.8. Corrected titles for 13.5-5-A and 13.5-6-H.
13.5.2.1.4	RAI NA3 13.05.02.01-3, P-STGs from GTGs
	RAI NA3 13.05.02.01-4, P-SWG re: EOPs and P-STGs
	Editorial correction.
13.5.2.2.1	RAI NA3 13.05.02.01-1, Management of Radioactive Waste
13.7, 13.7-202	Deleted references to pending revision to 10 CFR 26.
Table 13.4-201	Corrected entries in the Section column.
	Deleted the reference to a construction test program in Item 19.
	Consolidated snubber testing and inspection information into new item 20.
14.2.1.4, 14.2.7, 14.2.9, 14.2.9.1.1, 14.2.9.2.1	Changed supplements from STD to site-specific. Added reference to Initial Test Program implementation milestones. Clarified treatment of startup test procedures. Editorial changes.
14.2.2.1, 14AA	RAI NA3 14.02-3, Initial Test Program Administrative Document
14.2.8.1.36	RAI NA3 14.02-1, Initial Plant Test - Switchyard Components
14.2.9.1.4	RAI NA3 14.02-1, Initial Plant Test - Switchyard Components
14.3.8, 14.3.9, 14.3-1-A	Defined EP-ITAAC. Updated to align with DCD R5 changes.
14AA	RAI NA3 14.02-3, Initial Test Program Administrative Document
14AA.2.2.10	Consolidated multiple IRB names to FSRC.
	Added alternated IRB titles.
14AA.3.4	RAI NA3 14.02-7, Subsection 14.AA.3.4 - License Amendment
17.0, 17.2, 17.2.1, 17.3, 17.3.1, 17.4.10, 17.5	Changed supplements from STD to site-specific. Added reference to Operational QA Program implementation milestones.
17.5	Editorial change.
17.6.3	Deleted incorrect cross-referenced sections.

Revision 1 *(continued)*

Section	Changes
17AA	RAI NA3 17.05-4, QAPD Organization Charts; RAI NA3 17.05-5, Correct CFR Citation to 10 CFR 52.79(a)(27); & RAI 17.05-6, Commitment to RG 1.137
19.5, 19AA	RAI NA3 19-1, PRA and Severe Accident Evaluation (Internal Flooding) & RAI NA3 19-2, PRA and Severe Accident Evaluation (Site-Specific)

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Acronyms/Abbreviations/Initialisms

ADG	ancillary diesel generator
ALARA	as low as reasonably achievable
ALOHA	Areal Locations of Hazardous Atmospheres
AMC	Annual Maintenance Cost
ARD	Automatic Ringdown Circuits
ASCE	American Society of Civil Engineers
BISI	Bypass and Inoperable Status Indicator
BOP	Balance of Plant
bpf	blows per foot
BR	breathing rates
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CB	Control Building
CBR	California Bearing Ratio
CFR	Code of Federal Regulations
CIRC	Circulating Water System
CNO	chief nuclear officer
COL	Combined License
COLA	COL Application
CONAVS	Contaminated Area HVAC Subsystem
COVRERP	Virginia's Radiological Emergency Response Plan
CPT	cone penetrometer tests
CRF	Capital Recovery Factor
CS&TS	Condensate Storage and Transfer System
CSDRS	Certified Seismic Design Response Spectra
CST	Condensate Storage Tank
DBA	design basis accident
DC	Design Certification
DCD	Design Control Document
DCEM	Direct Cost of Equipment and Materials
DLC	Direct Labor Cost
D-RAP	design reliability assurance program
DTPG	defined test plan group
EAB	exclusion area boundary
EC	energy conservation
ECL	effluent concentration limit
EHC	electro-hydraulic control
EIS	Environmental Impact Statement
ENS	Emergency Notification System
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EPC	Engineer, Procure, and Construct
EQD	Equipment Qualification Document

Acronyms/Abbreviations/Initialisms

ERDS	Emergency Response Data Systems
ESP	Early Site Permit
ESPA	ESP Application
ETR	energy transfer ratio
FAC	flow accelerated corrosion
FES	Final Environmental Statements
FFD	Fitness for Duty
FIRS	foundation input response spectra
FMG	failure mode group
FOAK	first of a kind
FPS	Fire Protection System
FS	factor of safety
fps	feet per second
FSRC	Facilities Safety Review Committee
FWSC	Firewater Service Complex
gal	gallon
GE	General Electric
GEH	GE-Hitachi Nuclear Energy Americas, LLC
GIS	Geographic Information System
GMRS	ground motion response spectra
gpd	gallons per day
gpm	gallons per minute
GTG	Generic Technical Guidelines
GW	gigawatts
GWG	generic writer's guide
HCLPF	High Confidence Low Probability of Failure
HFE	Human Factors Engineering
HCLPFs	High Confidence Low Probability of Failure
HP	high-pressure
HPM	Human Performance Monitoring
HSI	Human System Interface
HWCS	Hydrogen Water Chemistry System
I&C	instrumentation and control
IBC	International Building Code
ICF	Indirect Cost Factor
IC/PCC	Isolation Condenser/Passive Containment Cooling
ICRP	International Commission on Radiation Protection
IDLH	immediately dangerous to life or health
IE	Inspection and Enforcement (NRC)
IRB	Independent Review Body
ISFSI	independent spent fuel storage installation
ISI	inservice inspection
IST	inservice testing

Acronyms/Abbreviations/Initialisms

JIT	just in time
JPM	job performance measures
JTG	Joint Test Group
ksf	kips per square foot
ksi	kips per square inch
LCCF	Labor Cost Correction Factor
LCO	limiting conditions for operation
LFL	lower flammability limit
LLD	lower limit of detection
LOPP	Loss of Preferred Power
LP	low-pressure
LWMS	Liquid Waste Management System
M&TE	measuring and test equipment
MCR	main control room
MCVP	main condenser vacuum pump
MEI	maximally exposed individual
min	minute
MOV	motor-operated valve
mph	miles per hour
MR	Maintenance Rule
msl	mean sea level
MWC	maximum water conservation
MWe	megawatts electric
MWS	Makeup Water System
NANIC	North Anna Nuclear Information Center
NAPS	North Anna Power Station
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NERC	North American Electric Reliability Corporation
NESC	National Electrical Safety Code
NPHS	normal plant heat sink
NRC	United States Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OATC	Operator-At-The Controls
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
ODEC	Old Dominion Electric Cooperative
OJT	on-the-job training
OSC	Operational Support Center
P&ID	piping and instrument diagrams
pcf	pounds per cubic foot
PCP	Process Control Program

Acronyms/Abbreviations/Initialisms

PCTMS	Plant Cooling Tower Makeup System
PGP	procedures generation package
PMF	probable maximum flood
PMP	probable maximum precipitation
PP	pocket penetrometer
ppm	parts per million
PSI	preservice inspection
P-STG	plant-specific technical guideline
PST	preservice test
PSWS	Plant Service Water System
PWSS	Pretreated Water Supply System
QA	quality assurance
QC	quality control
QAPD	Quality Assurance Program Description
RB	Reactor Building
RB/FB	Reactor Building/Fuel Building
RCCWS	Reactor Component Cooling Water System
RCS	reactor coolant system
RCTS	resonant column torsional shear
REPAVS	Refueling and Pool Area HVAC Subsystem
RG	Regulatory Guide
RO	reactor operator
RP	radiation protection
RQD	rock quality designation
RPT	radiation protection technician
RT	radiography techniques
RTNSS	Regulatory Treatment of Non-Safety Systems
RTO	Regional Transmission Organization
SACTI	Seasonal/Annual Cooling Tower Impact (computer code)
scfm	standard cubic feet per minute
scfw	standard cubic feet per week
SCG	Startup Controlling Group
SDG	standby diesel generator
SM	silty sand
SRO	senior reactor operator
SRP	Standard Review Plan
SNS	Station Nuclear Safety
SOV	solenoid-operated valve
SPT	standard penetration test
SS	site-specific
SSAR	Site Safety Analysis Report (ESPA Part 2)
SSCs	structures, systems, and components
SSE	Safe Shutdown Earthquake

Acronyms/Abbreviations/Initialisms

STA	Shift Technical Advisor
SUNSI	sensitive unclassified non-safeguards information
SWMB	Storm Water Management Basin
SWR	Service Water Reservoir
SWS	Station Water System
SWST	station water storage tank
TAC	Total Annual Cost
TBE	Turbine Building Air Exhaust Subsystem
TBVS	Turbine Building HVAC System
TCCWS	Turbine Component Cooling Water System
TGS	Turbine Generator Set
UAT	unit auxiliary transformer
UFL	upper flammability limit
UFSAR	Updated Final Safety Analysis Report
USCS	Unified Soil Classification System
UHS	ultimate heat sink
UT	ultrasonic techniques
V&V	verification and validation
VDH	Virginia Department of Health
V_p	compression wave velocity
V_s	shear wave velocity
VHRA	very high radiation area
WHTF	Waste Heat Treatment Facility

FINAL SAFETY ANALYSIS REPORT

Chapter 1 Introduction and General Description of Plant

1.1 Introduction

This section of the ESBWR Design Control Document (DCD), i.e., the referenced DCD, is incorporated by reference with the following departures and/or supplements.

1.1.1 Format and Content

NAPS SUP 1.1-1

1.1.1.1 10 CFR 52 and Regulatory Guide 1.206

This FSAR was developed to comply with the content requirements of 10 CFR 52.79, and to the extent feasible, the content and format requirements contained in Regulatory Guide (RG) 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).” See Table 1.9-203, Conformance With the FSAR Content Guidance In RG 1.206. If the information requested by RG 1.206 is not needed (e.g., because it is already provided in the DCD or is located elsewhere in the FSAR), the table specifies the location of the information.

Section C.III.6 of RG 1.206 addresses referencing a design certification (DC) application rather than a certified design. The existing DC rules (10 CFR 52 appendices) require that a Combined Operating License Application (COLA) that references a certified design include a plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for the ESBWR design, as modified and supplemented by the applicant’s exemptions and departures. Where necessary to present additional information, new sections were added following the logical structure of the ESBWR generic DCD.

1.1.1.2 Standard Review Plan

As required by 10 CFR 52.79(a)(41), an evaluation of the facility for conformance with the acceptance criteria contained in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants LWR Edition,” in effect six months prior to submittal of the COLA was performed. This evaluation determined that this FSAR contains no unacceptable deviations from the acceptance criteria given in the applicable portions of the SRP. Where necessary, Table 1.9-201, Conformance with Standard Review Plan, provides a summary of any differences from the SRP acceptance criteria, along with a justification for an exception to a criterion or a Branch Technical Position (BTP); or the table identifies the applicable FSAR section(s) that addresses a difference.

1.1.1.3 Tables and Figures

Tabulations of data are designated “tables.” Each is identified by the section number followed by a number (for example, Table 1.9-204 would be an FSAR table in Section 1.9.) The use of the “200” series for FSAR table numbers distinguishes FSAR tables from DCD tables. If a table from the DCD is referenced in the FSAR text, it is denoted as such, for example “DCD Table 4.1-1.” If a table from the DCD or Early Site Permit Application (ESPA) was revised for use in the FSAR, the original DCD or ESPA table number is appended with an “R;” for example, if “DCD Table 4.2-1” was revised, it would have become “Table 4.2-1R.” Tables are located at the end of the section immediately following the text.

Drawings, pictures, sketches, curves, graphs, and engineering diagrams identified as “figures” are numbered using the section number followed by a number (for example, Figure 2.1-201 would be an FSAR figure in Section 2.1). The use of the “200” series for FSAR figure numbers distinguishes FSAR figures from DCD or ESPA figures. If a figure from the DCD or ESPA is referenced in the FSAR text, it is denoted as such; for example “DCD Figure 4.1-1.” If a figure from the DCD or ESPA was revised for use in the FSAR, the original DCD or ESPA figure number was appended with an “R;” for example, if “DCD Figure 4.2-1” was revised, it would have become “Figure 4.2-1R.” Figures are located at the end of the applicable section following the tables.

1.1.1.4 Numbering of Pages

Text pages are numbered sequentially within each chapter (for example, Page 1-4 is the fourth page of Chapter 1).

1.1.1.5 Proprietary and Security-Related Sensitive Unclassified Non-Safeguards Information (SUNSI)

Proprietary information and SUNSI[1] is withheld from public disclosure and therefore not included in the public version of the FSAR. SUNSI included in the non-public version of the FSAR is appropriately indicated.

1.1.1.6 Acronyms

In addition to the summary list of acronyms in the FSAR frontmatter, acronyms are defined at their first occurrence in FSAR text.

1.1.1.7 Incorporation by Reference

10 CFR 52.79 states in part that, "The final safety analysis report need not contain information or analyses submitted to the Commission in connection with the design certification, provided, however, that the final safety analysis report must either include or incorporate by reference the standard design certification final safety analysis report and must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the site characteristics fall within the site parameters specified in the design certification." Therefore, because this COLA references the ESBWR DC application, this FSAR incorporates the ESBWR DCD by reference, with the departures presented in COLA Part 7, and with supplemental information, as appropriate (see Section 1.1.1.10). References in this FSAR to the DCD should be understood to mean the ESBWR DCD, Tier 2, submitted by GE-Hitachi Nuclear Energy Americas LLC (GEH), as Revision 5.

1.1.1.8 Departures from the Standard Design Certification (or Application)

A departure is a plant-specific "deviation" from design information in a standard DC rule or, consistent with Section C.III.6 of RG 1.206, from design information in a DC application.

10 CFR 52 clarifies that Tier 2 information in a standard DC rule does not include conceptual design information (CDI) and per Section C.III.6 of RG 1.206, Tier 2 information in a standard DC application does not

1. Any information which, if lost, misused, modified, or accessed without authorization, can reasonably be foreseen as causing harm to the public interest, the commercial or financial interest of the entity or individual to whom the information pertains, the conduct of NRC and Federal programs, or the personal privacy of individuals. SUNSI has been organized into the following seven groups:
   - Allegation information
   - Investigation information
   - Security-related information
   - Proprietary information
   - Privacy Act information
   - Federal, State, Foreign Government, and international agency information
   - Sensitive internal information

include CDI. Therefore, replacement or revision of CDI does not constitute a departure. Additionally, information addressing combined license (COL) information/holder items and supplemental information (see Section 1.1.1.10) that does not change the intent or meaning of the ESBWR DCD text is not considered a departure from the ESBWR DCD.

NAPS SUP 1.1-2

1.1.1.9 Referencing of ESPA Information

As with the DCD, the FSAR incorporates by reference the North Anna ESPA SSAR, Revision 9, with certain variances and/or supplements (see Section 1.1.1.10). A variance is a plant-specific deviation from one or more of the site characteristics, design parameters, or terms and conditions of an ESP or from the SSAR. A variance to an ESP is analogous to a departure from a standard DC.

SSAR Chapter 1 is incorporated by reference for historical purposes as an appendix to this chapter.

1.1.1.10 Supplements

Supplements fall into one of the following categories (see Table 1.1-201 for definitions of categories unless noted otherwise):

- COL Item
- Conceptual Design Information
- ESP COL Action Item
- ESP Permit Condition
- ESPA SSAR Correction
- Supplemental Information (see definition below)

Supplemental information is FSAR information that includes information not related to COL Items, departures, variances, conceptual design, ESPA corrections, or permit conditions (see Table 1.1-201 for definition of terms); or is information to demonstrate that the design of the facility falls within the site characteristics and design parameters specified in the DCD.

1.1.1.11 Left Margin Annotations

FSAR sections are annotated in the left margin with information that identifies: 1) the reason the information is being provided and, as applicable, 2) whether the information is standard (identical) for any ESBWR application, or specific to the COLA for a particular plant.

The annotations and their definitions are listed in Table 1.1-201.

1.1.1.12 Tense

Because this FSAR is a licensing basis document that will control plant design and operations after the COL is issued, the FSAR is generally written in the present tense. Thus, plant design and configuration are described in the present tense although the plant is not yet built. Similarly, programs, procedures, and organizational matters are generally described in the present tense although such descriptions may not yet be implemented. Accordingly, the use of the present tense in this FSAR should be understood as describing the plant, programs and procedures, and organization as they will exist when in place, and not as a representation that they are already in place.

1.1.2 General Description

1.1.2.1 ESBWR Standard Plant Scope

Replace the last sentence with the following.

NAPS CDI The orientation of the principal plant structures for Unit 3 is shown in Figure 2.1-201.

Add the following at the end of this section.

NAPS SUP 1.1-2 The ESBWR standard plant scope is discussed in DCD Section 1.1.2.1. In addition to the buildings and structures within the scope of the ESBWR standard plant, the plant includes an intake structure for plant makeup water, normal power heat sink and auxiliary heat sink cooling towers, a sewage treatment plant, water treatment facilities, storage tanks for water and fuel oil, a switchyard and other site support systems and structures necessary to support the operation and maintenance of the facility.

1.1.2.2 Type of License Request

Add the following at the end of this section.

NAPS SUP 1.1-3 This application by Virginia Electric and Power Company (Dominion) and the Old Dominion Electric Cooperative (ODEC) is for a combined construction permit and operating license, i.e., COL under Section 103 of the Atomic Energy Act, for the third nuclear power plant to be located on the existing North Anna Power Station (NAPS) site in Louisa County, Virginia. This COLA references a DC application for an ESBWR (consistent with Section C.III.6 of RG 1.206) and the Early Site Permit (ESP) for the NAPS site. The third unit is designated North Anna Unit 3 (Unit 3).

1.1.2.4 Description of Location

Add the following at the end of this section.

NAPS SUP 1.1-4

SSAR Section 2.1.1.1 is incorporated by reference with no departures or supplements.

1.1.2.7 Rated Core Thermal Power

Replace the last three sentences of this section with the following.

NAPS COL 1.1-1-A

Unit 3 operates at an estimated gross electrical power output at rated power of approximately 1594 MWe (as shown in DCD Section 10.1). The estimated net electrical power output, which is dependent on site ambient conditions, the normal plant heat sink (NPHS) operation controls, and station electrical loads, is between approximately 1425 MWe and 1510 Mwe.

NAPS SUP 1.1-5

1.1.2.8 Schedule

Key milestones associated with the estimated schedule for the completion of construction and the beginning of commercial operation are as follows.

Milestone	Estimated Schedule Date
Potential Safety-Related Construction Start	2012
Commercial Operation	2017

1.1.3 COL Unit-Specific Information

1.1-1-A Establish Rated Electrical Output

NAPS COL 1.1-1-A

This COL Item is addressed in Section 1.1.2.7.

NAPS SUP 1.1-1

Table 1.1-201 Left Margin Annotations

FSAR Component	Margin Annotation	Definition and Use
Standard Departure	STD DEP X.Y.Z -#	FSAR information that departs from the generic DCD and is common for all parallel applicants; i.e., the departure and discussion of the departure are identical for all applicants of the ESBWR technology. Each Standard Departure is numbered based on the applicable section down to the X.Y.Z level, e.g.: STD DEP 9.2-1, or STD DEP 9.2.1-1.
Plant-Specific Departure	(PLANT) DEP X.Y.Z-#	FSAR information that departs from the generic DCD and is plant-specific; i.e., the departure and discussion of the departure are not identical for all applicants of the ESBWR technology. Each Plant-Specific Departure is numbered based on the applicable section down to the X.Y.Z level, e.g.: NAPS DEP 9.2-1, or NAPS DEP 9.2.1-1.
Standard COL Item	STD COL X.Y-#-A or STD COL X.Y-#-H	FSAR information that addresses a DCD COL Item that is common for all parallel applicants; i.e., the response to and discussion of the DCD COL Item are identical for all applicants of the ESBWR technology. Each Standard COL Item is numbered as identified in ESBWR DCD Table 1.10-1. The –A refers to a COL Applicant item while the –H refers to a COL Holder item.
Plant-Specific COL Item	(PLANT) COL X.Y-#-A or (PLANT) COL X.Y-#-H	FSAR information that addresses a DCD COL Item that is plant-specific; i.e., the response to the COL Item is not a Standard COL Item for parallel applicants. Each Plant-Specific COL Item is numbered as identified in the ESBWR DCD (see STD COL above).

NAPS SUP 1.1-1

Table 1.1-201 Left Margin Annotations

FSAR Component	Margin Annotation	Definition and Use
Standard Conceptual Design Information	STD CDI	A Conceptual Design Information designation is used to identify FSAR information that replaces Conceptual Design Information in the DCD, in whole or in part. Replacement and supplemental Conceptual Design Information is generally plant-specific; however, for conceptual design that is generic for all applications the annotation for standard (STD) is used, STD CDI.
Plant Specific Conceptual Design Information	(PLANT) CDI	A Conceptual Design Information designation is used to identify FSAR information that replaces Conceptual Design Information in the DCD, in whole or in part. Plant specific replacement and supplemental Conceptual Design Information uses the annotation (PLANT) CDI, e.g., NAPS CDI.
Standard Supplemental Information	STD SUP X.Y-#	Supplemental FSAR information that is identical for all parallel applicants; i.e., the supplemental information is identical for all applicants of the ESBWR technology. Each Standard Supplemental Information designation is numbered based on applicable section down to the X.Y level, e.g., STD SUP 10.4-1.
Plant-Specific Supplemental Information	(PLANT) SUP X.Y-#	Supplemental FSAR information that is plant-specific (not standard). Each Plant Specific Supplemental Information designation is numbered based on applicable section down to the X.Y level, e.g., NAPS SUP 10.4-1.
ESP COL Item	(PLANT) ESP COL X.Y-#	ESP COL Action items identify matters that an applicant for a construction permit or operating license addresses in a COLA. An ESP COL Item designation is used to identify FSAR information that addresses an ESP COL Action Item. Responses to all ESP COL Action Items are assumed to be plant-specific. An ESP COL Action Item is numbered as identified in the applicable ESP; e.g., NAPS ESP COL 2.4-2.

NAPS SUP 1.1-1

Table 1.1-201 Left Margin Annotations

FSAR Component	Margin Annotation	Definition and Use
ESP Permit Condition	(PLANT) ESP PC #	ESP Permit Conditions are requirements to take certain actions as specified in that permit. An ESP Permit Condition designation is used to identify FSAR information that addresses an ESP Permit Condition. Responses to all ESP Permit Conditions are assumed to be plant-specific. An ESP Permit Condition is numbered as identified in the applicable ESP; e.g., NAPS ESP PC 3.E(1).
ESP Variance	(PLANT) ESP VAR X.Y.Z-#	A request for an ESP Variance is a request for deviation from one or more site characteristics, design parameters, or terms and conditions of the ESP; or from the SSAR. Each ESP Variance is numbered based on the applicable section down to the X.Y.Z level, e.g., NAPS ESP VAR 2.4-1.
Early Site Permit Safety Analysis Report Corrections	ESP COR	Corrections to the information provided in the ESP safety analysis report in order to ensure that the information is complete and accurate for FSAR.

1.2 General Plant Description

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

1.2.2.11.4 Main Turbine

Delete the second sentence of the first paragraph and replace the first sentence of the first paragraph with the following.

STD CDI The main turbine has one high-pressure (HP) turbine and three low-pressure (LP) turbines.

1.2.2.11.7 Main Condenser

Delete the second sentence of the third paragraph and replace the first sentence of the third paragraph with the following.

STD CDI The main condenser is a multi-pressure, triple-shell unit.

1.2.2.12.7 Plant Service Water System

Delete the last sentence of the first paragraph; delete the second and third sentences of the second paragraph; and revise the first sentence of the second paragraph as follows.

NAPS CDI The PSWS mechanical draft plume abated cooling towers are used to reject the heat removed from Reactor Component Cooling Water System (RCCWS) and Turbine Component Cooling Water System (TCCWS).

1.2.2.12.13 Hydrogen Water Chemistry System

Replace this section with the following.

STD CDI The Hydrogen Water Chemistry System (HWCS) consists of hydrogen and oxygen supply systems to inject hydrogen in the feedwater and oxygen in the offgas, plus monitoring systems to track the effectiveness of the system.

1.2.2.12.15 Zinc Injection System

Replace this section with the following.

STD CDI The Zinc Injection System is not utilized.

1.2.2.12.16 **Freeze Protection**

Replace this section with the following.

STD CDI

Freeze protection is incorporated at the individual system level using insulation and heat tracing for all external tanks and piping that may freeze during winter weather.

1.2.2.16.10 **Other Building Structures**

Replace the fifth paragraph with the following.

NAPS CDI

Other facilities include the Service Building, Water Treatment Building, Administration Building, Training Center, Sewage Treatment Plant, and hot machine shop. These are all of conventional size and design, and in some cases may be shared with other units at the same site.

STD SUP 1.2-1

1.2.2.19 **Modular Construction Techniques and Plans**

To the extent practical, modular construction techniques that have been applied during ABWR construction projects will be adapted and/or modified for use during ESBWR construction. Modularization reviews will be performed to develop a plan for bringing the ABWR experience into the ESBWR. Once completed, the results of the modularization reviews will be used as guidance to develop the detailed design of the areas affected by modularization.

1.3 Comparison Tables

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

Add the following at the end of this section.

NAPS COL 1.3-1-A

There are no updates to DCD Table 1.3-1 based on unit-specific information.

1.3.1 COL Information

1.3-1-A **Update Table 1.3-1**

NAPS COL 1.3-1-A

This COL item is addressed in Section 1.3.

1.4 Identification of Agents and Contractors

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

NAPS SUP 1.4-1

1.4.1 Dominion

Dominion and ODEC are the applicants for the COL, and Dominion will be the licensee authorized to construct and operate Unit 3. Dominion is therefore responsible for making each of the key project decisions, including the ultimate decision on whether to build a new nuclear power plant, and would be the plant operator.

Dominion has selected GE-Hitachi Nuclear Energy Americas, LLC (GEH) as its primary contractor for the design of the unit, and Bechtel Power Corporation (Bechtel) as the primary contractor for site engineering. Dominion has responsibility for the operation of the unit. The following sections provide information on the experience and qualifications of the aforementioned agents and contractors as well as the division of responsibility between Dominion and its agents and contractors.

1.4.2 GE-Hitachi Nuclear Energy Americas, LLC (GEH)

GEH is responsible for developing the complete standard plant for the ESBWR necessary to obtain a DC from the NRC, supporting preparation of the COL application, and activities to support deployment of the ESBWR on the North Anna site. GEH, established in June 2007, is a business alliance of GE and Hitachi's respective nuclear businesses, established to serve the global nuclear industry.

DCD Table 1.4-1 lists the commercial nuclear reactors that were completed by GE or are under construction by GEH. For 50 years, GE provided advanced technology for nuclear energy. GE developed breakthrough light water technology in the mid-1950s: the Boiling Water Reactor (BWR). Since then, GE developed nine evolutions of BWR technology, including the first operational advanced light water design in the world, the ABWR, and culminating in its latest generation of design, the ESBWR. All of GE's nuclear technology has been transferred to GEH. There are 67 plants operating worldwide utilizing GEH designs with an operating capacity of over 59 GW, including 36 BWR plants in North America. Various subcontractors are supporting GEH.

1.4.2.1 Construction of the Turbine Island and Nuclear Island

The contractors for the construction of the turbine island and the nuclear island have not yet been selected. The turbine island and the nuclear island together represent the power block. The contractor for the construction of the turbine island will be responsible for the erection and delivery of the turbine building, the electric building, and the contents of each building. The contractor for the construction of the nuclear island will be responsible for the erection and delivery of the reactor and fuel building, the control building, the hot machine shop, the radwaste building, and the contents of each building. Each contractor will be selected based on their historical work in the nuclear industry, ongoing nuclear business, ability to deliver integrated engineering and construction services, and available resources.

1.4.3 Bechtel Power Corporation

Bechtel is responsible for the engineering and licensing support of the COLA, and for site engineering of facilities and utilities outside of the plant power block.

Founded in 1898, Bechtel is one of the world’s premier engineering, construction, and project management companies. Privately owned with headquarters in San Francisco, Bechtel has 40 offices around the world and 40,000 employees. Bechtel has a history of supporting the nuclear power industry, beginning with the construction in 1950 of the EBR-1 reactor. Since then, Bechtel has constructed more than 60 GWe of nuclear power capacity worldwide. Various subcontractors are supporting Bechtel.

1.4.4 Other Contractors

In addition to the major contractors listed above, contractual relationships were established with several specialized consultants to assist in developing the COLA. Other subcontractors may be added as the need arises.

1.4.4.1 Tetra Tech NUS, Inc.

Tetra Tech NUS, Inc. conducted new and significant information reviews for the Environmental Report and prepared several sections of the Environmental Report, including the ecological description of the site and vicinity, environmental impacts of construction, and plant cooling system impacts on terrestrial and aquatic ecosystems. Tetra Tech NUS, Inc. also provided general National Environmental Policy Act (NEPA) consultation.

1.4.4.2 MACTEC Engineering and Consulting, Inc.

MACTEC Engineering and Consulting, Inc. performed geotechnical field investigations and laboratory testing in support of Chapter 2. That effort included performing standard penetration tests; obtaining core samples and rock cores; performing cone penetrometer tests, cross-hole seismic tests, and laboratory tests of soil and rock samples; installing ground water observation wells; and preparing a data report.

1.4.4.3 Risk Engineering, Inc.

Risk Engineering, Inc. performed probabilistic seismic hazard assessments and related sensitivity analyses in support of Chapter 2. These assignments included sensitivity analyses of seismic source parameters and updated ground motion attenuation relationships, development of updated Safe Shutdown Earthquake (SSE) ground motion values, and preparation of the related sections.

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 Incorporated by Reference

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

Add the following paragraph at the end of this section.

NAPS SUP 1.6-1 Table 1.6-201 lists topical reports not included in DCD Section 1.6 that are incorporated in whole or in part by reference in the FSAR.

NAPS SUP 1.6-1

Table 1.6-201 Referenced Topical Reports

Report No.	Title	Section
NEI 06-13A	Nuclear Energy Institute, “Technical Report on Template for an Industry Training Program Description,” NEI 06-13A, Revision 1, March 2008	13BB
NEI 06-14A	Nuclear Energy Institute, “Quality Assurance Program Description,” NEI 06-14A, Revision 4, July 2007	17.5
NEI 07-02A	Nuclear Energy Institute, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed under 10 CFR Part 52,” NEI 07-02A, March 2008	17.6
NEI 07-03	Nuclear Energy Institute, “Generic FSAR Template Guidance for Radiation Protection Program Description,” NEI 07-03, Revision 3, October 2007	12BB
NEI 07-08	Nuclear Energy Institute, “Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA),” NEI 07-08, Revision 0, September 2007	12AA
NEI 07-09	Nuclear Energy Institute, “Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description,” NEI 07-09, Revision 0, September 2007	11.5
NEI 07-10	Nuclear Energy Institute, “Generic FSAR Template Guidance for Process Control Program (PCP),” NEI 07-10, Revision 2, February 2008	11.4

1.7 Drawings and Other Detailed Information

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

1.7.1 Electrical, Instrumentation and Control Drawings

Add the following at the end of this section.

NAPS SUP 1.7-1 Table 1.7-201 supplements DCD Table 1.7-2 for those portions of the electrical system configuration drawings outside the scope of the DCD.

1.7.2 Piping and Instrumentation Diagrams

Add the following at the end of the first paragraph.

NAPS SUP 1.7-1 Table 1.7-202 supplements DCD Table 1.7-3 for those portions of the mechanical system configuration drawings outside the scope of the DCD.

Replace the last paragraph of this section with the following.

STD COL 1.7-1-H The final P&IDs used for construction will be available upon completion of the final design configuration. Design changes that result in revisions to the simplified diagrams will be incorporated in subsequent updates to this FSAR.

1.7.4 COL information

1.7-1-H Final Design Configuration Confirmation

STD COL 1.7-1-H This COL item is addressed in Section 1.7.2.

NAPS SUP 1.7-1

Table 1.7-201 Summary of Electrical System Configuration Drawings

Figure 8.2-201, 500/230 kV Switchyard Single-Line Diagram
Figure 8.2-202, 500/230 kV Switchyard Arrangement
Figure 8.2-203, Dominion Transmission Line Map

NAPS SUP 1.7-1

Table 1.7-202 Summary of Mechanical System Configuration Drawings

Figure 9.2-201, Plant Service Water System Simplified Diagram
Figure 9.2-202, Potable Water System Simplified Diagram
Figure 9.2-203, Sanitary Waste Discharge System Simplified Diagram
Figure 9.2-204, Station Water System - Plant Cooling Tower Makeup System (PCTMS)
Figure 9.2-205, Station Water System - Pretreated Water Supply System (PWSS)
Figure 9.5-201, Fire Protection System; Main Yard Loop
Figure 9.5-202, Fire Protection System Secondary Fire Pumps
Figure 9.5-203, Fire Protection System; Cooling Tower Yard Loop
Figure 10.4-201, Circulating Water Pumps
Figure 10.4-202, Dry Cooling Tower Array
Figure 10.4-203, Hybrid Cooling Tower

1.8 Interfaces with Standard Design

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

1.8.2 Identification of Balance of Plant Interfaces

Add the following paragraph after the first paragraph of this section.

STD CDI

The significant interface requirements for those systems that are beyond the scope of the DCD are identified in DCD Tier 1.

Delete the second sentence of the second paragraph of this section.

NAPS SUP 1.8-1

1.8.3 Verification of Site Parameters

Chapter 2 provides information demonstrating that the site characteristics fall within the ESBWR site parameters specified in the referenced certified design.

Chapter 2 also provides information demonstrating that the design of the facility falls within the site characteristics and bounding design parameters for the ESP (Reference 1.8.202).

NAPS SUP 1.8-2

1.8.4 COL Information Items and Permit Conditions

Section 1.10 identifies specific FSAR sections that address the COL information items from the referenced certified design, and COL Action Items and Permit Conditions from the ESP.

NAPS SUP 1.8-3

1.8.5 Generic Changes and Departures from the Referenced Certified Design

There are no generic changes or departures from the referenced certified design. (Reference Table 1.8-201)

NAPS SUP 1.8-4

1.8.6 Variances from the ESP and ESPA SSAR

Requests for variances from the ESP and SSAR comply with the requirements of 10 CFR 52.39 and 10 CFR 52.93. Variances are listed in Table 1.8-202, along with the section of the FSAR in which each is discussed. These variances are described and evaluated in COLA Part 7.

NAPS SUP 1.8-5

1.8.7 Conceptual Design Information

The referenced DCD includes conceptual design information (CDI) for certain systems, or portions of systems, that are outside the scope of the standard plant design. Table 1.8-203 identifies systems for which either the CDI in the DCD is adopted as the actual system design information, or the CDI in the DCD is replaced with site-specific design information, along with cross references to FSAR sections where the CDI is treated. Where there are differences between the conceptual design and the actual design, these differences have been evaluated. The evaluations have concluded that there are no impacts on the safety evaluations provided in the referenced certified design.

NAPS SUP 1.8-6

1.8.8 Probabilistic Risk Assessment

Site- and plant-specific information, including site meteorological data and site-specific population distribution, plant-specific design information that replaced conceptual design information described in the DCD, and the departures listed in Section 1.8.5, were reviewed with respect to the design certification PRA. The conclusion, which is documented in Section 19.5, is that there is no significant change from the certified design PRA.

1.8 References

1.8.201 [Deleted]

1.8.202 Early Site Permit (ESP) for the North Anna ESP Site, No. ESP-003, U.S. Nuclear Regulatory Commission, November 2007.

NAPS SUP 1.8-3

Table 1.8-201 Departures from the Referenced Certified Design

Number	Subject	FSAR Section
None		

NAPS SUP 1.8-4

Table 1.8-202 Variances from the ESP and ESPA SSAR

Number	Subject	FSAR Location
NAPS ESP VAR 2.0-1a-l	Long-Term Dispersion Estimates (χ/Q and D/Q)	Section 2.3.5, Table 2.0-201
NAPS ESP VAR 2.0-2	Hydraulic Conductivity	Section 2.4.12.1.2, Table 2.0-201
NAPS ESP VAR 2.0-3	Hydraulic Gradient	Section 2.4.12.1.2, Table 2.0-201
NAPS ESP VAR 2.0-4	Vibratory Ground Motion	Section 2.5.2.5, Table 2.0-201
NAPS ESP VAR 2.0-5a-h	Distribution Coefficients (K_d)	Table 2.0-201
NAPS ESP VAR 2.0-6	DBA Source Term Parameters and Doses	Table 2.0-201
NAPS ESP VAR 2.0-7a-b	Coordinates and Abandoned Mat Foundations	Table 2.0-201
NAPS ESP VAR 2.4-1	Void Ratio, Porosity, and Seepage Velocity	Section 2.4.12.1.2
NAPS ESP VAR 2.4-2	NAPS Water Supply Well Information	Table 2.4-17R
NAPS ESP VAR 2.5-1	Stability of Slopes	Section 2.5.5
NAPS ESP VAR 2.5-2	Engineered Fill	Section 2.5.1.2.3.k Section 2.5.4.5.3
NAPS ESP VAR 12.2-1	Gaseous Pathway Doses	Section 12.2.2.2.6, Table 12.2-18bR
NAPS ESP VAR 12.2-2	[Deleted]	
NAPS ESP VAR 12.2-3	Annual Liquid Effluent Releases	Section 12.2.2.4.6, Table 12.2-19bR
NAPS ESP VAR 12.2-4	Existing Units' and Site Total Doses	Table 12.2-203

NAPS SUP 1.8-5

Table 1.8-203 Conceptual Design Information (CDI)

Item in DCD	CDI in DCD adopted as actual design	CDI in DCD replaced with actual design	Evaluation	FSAR Section
1.1.2.1 ESBWR Standard Plant Scope Figure 1.1-1 ESBWR Standard Plant General Site Plan		X	Site-specific plan general site plan provided	1.1.2.1 Figure 2.1-201
1.2.2.11.4 Main Turbine	X		Conceptual turbine type selected as site-specific design	1.2.2.11.4
1.2.2.11.7 Main Condenser	X		Conceptual condenser type selected as site-specific design	1.2.2.11.7
1.2.2.12.7 Plant Service Water System		X	Site-specific design described	1.2.2.12.7
1.2.2.12.13 Hydrogen Water Chemistry Table 3.2-1 P73 Note 9.3.9 Hydrogen Water Chemistry		X	Hydrogen water chemistry option utilized	1.2.2.12.13 Table 3.2-1 9.3.9
1.2.2.12.15 Zinc Injection System Table 3.2-1 P74 Note 9.3.11 Zinc Injection System		X	Zinc Injection system not utilized	1.2.2.12.15 Table 3.2-1 9.3.11
1.2.2.12.16 Freeze Protection		X	Freeze protection incorporated for external tanks and piping that may freeze during winter weather	1.2.2.12.16
1.2.2.16.10 Other Building Structures		X	Site-specific buildings specified	1.2.2.16.10
1.8.2 Identification of BOP Interfaces	X		Not applicable	1.8.2
Appendix 3A Seismic Soil-Structure Interaction Analysis		X	Site-specific geotechnical data described in Chapter 2	Appendix 3A Chapter 2
Appendix 3A.2 ESBWR Standard Site Plan		X	Site-specific general site plan provided	Section 3A.2 Figure 2.1-201

NAPS SUP 1.8-5

Table 1.8-203 Conceptual Design Information (CDI)

Item in DCD	CDI in DCD adopted as actual design	CDI in DCD replaced with actual design	Evaluation	FSAR Section
9.2.1 Plant Service Water Table 9.2-2 Figure 9.2-1		X	Site-specific system description and design characteristics described	9.2.1 Table 9.2-201 Figure 9.2-201
9.2.3 Makeup Water System Table 9.2-9		X	Site-specific system description and design characteristics described	9.2.3 Table 9.2-202
9.2.4 Potable and Sanitary Water Systems		X	Site-specific system description and design characteristics described	9.2.4 Figure 9.2-202 Figure 9.2-203
9.2.10 Station Water System		X	Site-specific system description and design characteristics described	9.2.10 Table 9.2-203 Table 9.2-204 Figure 9.2-204 Figure 9.2-205
9.3.9 Hydrogen Water Chemistry System		X	Site-specific system description and design characteristics described	9.3.9
9.3.11 Zinc Injection System		X	Zinc Injection System not utilized	9.3.11

Table 1.8-203 Conceptual Design Information (CDI)

Item in DCD	CDI in DCD adopted as actual design	CDI in DCD replaced with actual design	Evaluation	FSAR Section
9A Appendix 9A Fire Hazards Analysis		X	Site-specific buildings specified. Site-specific Fire Zone drawings supplied.	9A Contents 9A.1 9A.3.1 9A.4.9 9A.5.9 9A.5.12 Figure 9A.2-33R Figures 9A.2-201 through 9A.2-206
10.4.5 Circulating Water System Table 10.4-3 Figure 10.4-1		X	Site-specific system description and design characteristics described	10.4.5.2.1 Table 10.4-201 Table 10.4-3R Figure 10.4-201 Figure 10.4-202 Figure 10.4-203

1.9 Conformance with Standard Review Plan and Applicability of Codes and Standards

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

1.9.1 Conformance with Standard Review Plan

Add the following paragraph at the end of this section.

NAPS COL 1.9-3-A

Table 1.9-201 evaluates conformance with the SRP sections and BTPs in effect six months prior to the submittal of the COLA. Table 1.9-201 does not re-address conformance with the SRP for those portions of the facility design included in the referenced certified design. Similarly, Table 1.9-201 does not re-address SSAR conformance with the applicable RS-002 sections.

In the table, the term "Conforms" means that no exception is being taken to the guidance in the SRP section/acceptance criteria as they apply to site-specific design information, operational aspects of the facility, or siting information in the FSAR that supplements the SSAR. The term "Not applicable" means that the SRP section/acceptance criteria do not apply to the ESBWR or Unit 3. Any differences with the SRP acceptance criteria are identified and justified, with references to the applicable FSAR section(s) that address the difference, as necessary.

1.9.2 Applicability to Regulatory Criteria

Add the following paragraphs at the end of this section.

NAPS COL 1.9-3-A

Division 1, 4, 5, and 8 Regulatory Guides

Table 1.9-202 evaluates conformance with Division 1, 4, 5, and 8 RGs in effect six months prior to the submittal of the COLA. Each issued Division 1 RG is evaluated. Issued Division 4, 5, and 8 RGs identified in the SRP, RG 1.206, or DCD Table 1.9-21 as COL responsibility are also evaluated. (Conformance with Division 4 RGs is also addressed in COLA Part 3, Section 1.4.) Table 1.9-202 does not re-address conformance with RGs for those portions of the facility design included in the referenced certified design. Similarly, Table 1.9-202 does not re-address SSAR conformance with the applicable RGs.

In the table, the term "Conforms" means that no exception is being taken to the guidance in the regulatory positions as they apply to site-specific design information, operational aspects of the facility, or siting information in the FSAR that supplements the SSAR. The term "Not applicable" means that the regulatory positions do not apply to the ESBWR or Unit 3.

Regulatory Guide 1.206

Table 1.9-203 evaluates conformance with the FSAR content guidance in RG 1.206. Where necessary, the table identifies the FSAR section where the required information is provided. In the table, the term "Conforms" means that the information called for in RG 1.206 is either: 1) already addressed in the DCD or SSAR; or 2) addressed by adding new information beyond that contained in the DCD or SSAR. The term "Not applicable" means that the information called for in RG 1.206 does not apply to the ESBWR or Unit 3.

Table 1.9-203 evaluates conformance with RG 1.206, Section C.III.2, "Information Needed for a Combined License Application Referencing a Certified Design and an Early Site Permit." Section C.III.1, "Information Needed for a Combined License Application Referencing a Certified Design," and Section C.I, "Standard Format and Content of Combined License Applications for Nuclear Power Plants-Light-Water Reactor Edition," were also evaluated, as applicable, if portions of these sections were referenced or identified in RG 1.206, Section C.III.2, or Section C.III.1, respectively.

NAPS SUP 1.9-1

Industrial Codes and Standards

Table 1.9-204 identifies the Industrial Codes and Standards that are applicable to those portions of the Unit 3 design that are beyond the scope of the DCD or the SSAR, and to the operational aspects of the facility.

1.9.3 Applicability of Experience Information

Add the following after the first sentence of the section.

NAPS SUP 1.9-2

Table 1.9-205 lists NUREG and NUREG/CR reports cited in the FSAR.

Add the following paragraph at the end of this section.

Table 1.9-205 addresses operational experience information, as described in applicable NUREG reports, for those portions of the Unit 3 design and operation that are beyond the scope of the DCD. The comment column of Table 1.9-205 includes a reference to the applicable

FSAR section that provides further discussion of the operational experience.

1.9.4 COL Information

1.9-3-A SRP and Regulatory Guide Applicability

NAPS COL 1.9-3-A This COL Item is addressed in Sections 1.9.1 and 1.9.2.

NAPS COL 1.9-3-A

Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
1	Introduction and Interfaces	Initial Issuance	Mar-07	No Specific Acceptance Criteria	Conforms
2.0	Site Characteristics and Site Parameters	Initial Issuance	Mar-07	II.1, II.2, II.3, II.5	Not applicable
				II.4	Conforms
2.1.1	Site Location and Description	Rev. 3	Mar-07	II.1, II.2	Conforms
2.1.2	Exclusion Area Authority and Control	Rev. 3	Mar-07	II.1, II.2, II.3	Conforms
2.1.3	Population Distribution	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms
2.2.1–2.2.2	Identification of Potential Hazards in Site Vicinity	Rev. 3	Mar-07	II.1, II.2, II.3	Conforms
2.2.3	Evaluation of Potential Accidents	Rev. 3	Mar-07	II.1, II.2	Conforms
2.3.1	Regional Climatology	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9	Conforms
2.3.2	Local Meteorology	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
2.3.3	Onsite Meteorological Measurements Programs	Rev. 3	Mar-07	II.1, II.2, II.3	Conforms
2.3.4	Short Term Atmospheric Dispersion Estimates for Accident Releases	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
2.3.5	Long-Term Atmospheric Dispersion Estimates for Routine Releases	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms
2.4.1	Hydrologic Description	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms
2.4.2	Floods	Rev. 4	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10	Conforms
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	Rev. 4	Mar-07	II.1, II.2, II.3	Conforms
2.4.4	Potential Dam Failures	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
2.4.5	Probable Maximum Surge and Seiche Flooding	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms
2.4.6	Probable Maximum Tsunami Hazards	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8	Conforms
2.4.7	Ice Effects	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms
2.4.8	Cooling Water Canals and Reservoirs	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
2.4.9	Channel Diversions	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
2.4.10	Flooding Protection Requirements	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
2.4.11	Low Water Considerations	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms
2.4.12	Groundwater	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms

NAPS COL 1.9-3-A

Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
2.4.13	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	Rev. 3	Mar-07	II.1	Conforms. The relatively simple hydrogeologic conditions preclude the need to evaluate alternative conceptual models of the groundwater system. Alternative conceptual models of the more complex surface water system are evaluated to identify the bounding conditions.
				II.2, II.5	Conforms
				II.3	Conforms. Distribution coefficients conservatively assigned from literature values and compared to site-specific distribution coefficients.
				II.4	Conforms. There are no site-proximity hazards, seismic, or non-seismic events that would increase the radionuclide concentrations above the values reported in Section 2.4.13.
2.4.14	Technical Specifications and Emergency Operation Requirements	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms
2.5.1	Basic Geologic and Seismic Information	Rev. 4	Mar-07	II.1, II.2	Conforms
2.5.2	Vibratory Ground Motion	Rev. 4	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms
2.5.3	Surface Faulting	Rev. 4	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
2.5.4	Stability of Subsurface Materials and Foundations	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11, II.12	Conforms
2.5.5	Stability of Slopes	Rev. 3	Mar-07	II.1, II.2, II.3	Conforms
3.2.1	Seismic Classification	Rev. 2	Mar-07	II.1	Conforms
3.2.2	System Quality Group Classification	Rev. 2	Mar-07	II.1	Conforms
3.3.1	Wind Loadings	Rev. 3	Mar-07	II.1, II.2, II.3	Conforms
3.3.2	Tornado Loadings	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
3.4.1	Internal Flood Protection for Onsite Equipment Failures	Rev. 3	Mar-07	II.1, II.2	Conforms
3.4.2	Analysis Procedures	Rev. 3	Mar-07	II.1, II.2, II.3	Conforms
3.5.1.1	Internally Generated Missiles (Outside Containment)	Rev. 3	Mar-07	II.1, II.2	Conforms
3.5.1.2	Internally-Generated Missiles (Inside Containment)	Rev. 3	Mar-07	II.1, II.2	Conforms
3.5.1.3	Turbine Missiles	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms
3.5.1.4	Missiles Generated by Tornadoes and Extreme Winds	Rev. 3	Mar-07	II.1, II.2	Conforms
3.5.1.5	Site Proximity Missiles (Except Aircraft)	Rev. 4	Mar-07	II.1, II.2	Conforms
3.5.1.6	Aircraft Hazards	Rev. 3	Mar-07	II.1, II.2	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
3.5.2	Structures, Systems, and Components to be Protected from Externally-Generated Missiles	Rev. 3	Mar-07		Conforms
3.5.3	Barrier Design Procedures	Rev. 3	Mar-07	II.1, II.2	Conforms
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	Rev. 2	Mar-07	II.1, II.2, II.3	Conforms
3.6.3	Leak-Before-Break Evaluation Procedures	Rev. 1	Mar-07	II.1, II.2	Not applicable. ESBWR design does not rely on a Leak Before Break Evaluation.
3.7.1	Seismic Design Parameters	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
3.7.2	Seismic System Analysis	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11, II.12, II.13, II.14	Conforms
3.7.3	Seismic Subsystem Analysis	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11, II.12, II.13, II.14	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
3.7.4	Seismic Instrumentation	Rev. 2	Mar-07	II.1, II.2	Conforms
3.8.1	Concrete Containment	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
3.8.2	Steel Containment	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
3.8.4	Other Seismic Category I Structures	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8	Conforms
3.8.5	Foundations	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
3.9.1	Special Topics for Mechanical Components	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
3.9.2	Dynamic Testing and Analysis of Systems, Structures, and Components	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
3.9.3	ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures	Rev. 2	Mar-07	II.1, II.2, II.3	Conforms
3.9.4	Control Rod Drive Systems	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
3.9.5	Reactor Pressure Vessel Internals	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
3.9.6	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	Rev. 3	Mar-07	II.1, II.3, II.4, II.5, II.6	Conforms
				II.2	Not applicable. There are no safety related pumps.
3.9.7	Risk-Informed Inservice Testing	Rev. 0	Aug-98	II.A, II.B	Not applicable. Risk-informed inservice testing is not being used.
3.9.8	Risk-Informed Inservice Inspection of Piping	Rev. 0	Sep-03	II.1, II.2, II.3	Not applicable. Risk-informed inservice inspection of piping is not being used.
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	Rev. 3	Mar-07	II.1, II.2, II.3, II.5	Conforms
				II.4, II.6	Conforms
3.11	Environmental Qualification of Mechanical and Electrical Equipment	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11, II.12, II.13, II.14, II.15	Conforms
				II.16	Conforms
3.12	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports	Initial Issuance	Mar-07	II.A, II.B, II.C, II.D	Conforms
3.13	Threaded Fasteners - ASME Code Class 1, 2, and 3	Initial Issuance	Mar-07	II.1, II.2	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
BTP 3-1	Classification of Main Steam Components Other than the Reactor Coolant Pressure Boundary for BWR Plants	Rev. 2	Mar-07		Conforms
BTP 3-2	Classification of BWR/6 Main Steam and Feedwater Components Other than the Reactor Coolant Pressure Boundary	Rev. 2	Mar-07		Conforms
BTP 3-3	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	Rev. 3	Mar-07		Conforms
BTP 3-4	Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	Rev. 2	Mar-07		Conforms
4.2	Fuel System Design	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
4.3	Nuclear Design	Rev. 3	Mar-07	II.1, II.2, II.4	Conforms
				II.3	Conforms
4.4	Thermal and Hydraulic Design	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.8, II.9, II.10	Conforms
				II.7	Not applicable

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
4.5.1	Control Rod Drive Structural Materials	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
4.5.2	Reactor Internal and Core Support Structure Materials	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms
4.6	Functional Design of Control Rod Drive System	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8	Conforms
BTP 4-1	Westinghouse Constant Axial Offset Control (CAOC)	Rev. 3	Mar-07		Not applicable to the ESBWR
5.2.1.1	Compliance with the Codes and Standards Rule, 10 CFR 50.55a	Rev. 3	Mar-07	RG 1.26	Conforms
5.2.1.2	Applicable Code Cases	Rev. 3	Mar-07	RG 1.84, RG 1.147, RG 1.192	Conforms
5.2.2	Overpressure Protection	Rev. 3	Mar-07	II.1, II.2, II.5, II.6, II.7	Conforms
				II.3, & II.4	Not applicable to the ESBWR
5.2.3	Reactor Coolant Pressure Boundary Materials	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms. Acceptance Criterion II.3 is addressed in DCD Section 3.9.3.9.
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11	Conforms
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	Rev. 2	Mar-07	II.1, II.2	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
5.3.1	Reactor Vessel Materials	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
5.3.2	Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	Rev. 2	Mar-07	II.1, II.2, II.3	Conforms
5.3.3	Reactor Vessel Integrity	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8	Conforms
5.4	Reactor Coolant System Component and Subsystem Design	Rev. 2	Mar-07		Conforms
5.4.1.1	Pump Flywheel Integrity (PWR)	Rev. 2	Mar-07		Not applicable to the ESBWR
5.4.2.1	Steam Generator Materials	Rev. 3	Mar-07		Not applicable to the ESBWR
5.4.2.2	Steam Generator Program	Rev. 2	Mar-07		Not applicable to the ESBWR
5.4.6	Reactor Core Isolation Cooling System (BWR)	Rev. 4	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10	Conforms
5.4.7	Residual Heat Removal (RHR) System	Rev. 4	Mar-07	II.1, II.2, II.3, II.4	Conforms
5.4.8	Reactor Water Cleanup System (BWR)	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
5.4.11	Pressurizer Relief Tank	Rev. 3	Mar-07		Not applicable to the ESBWR

NAPS COL 1.9-3-A

Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
5.4.12	Reactor Coolant System High Point Vents	Rev. 1	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11, II.12, II.13, II.14	Conforms
5.4.13	Isolation Condenser System (BWR)	Initial Issuance	Mar-07	II.1, II.2, II.3, II.5, II.6, II.7, II.8, II.9, II.10, II.11, II.12	Conforms
				II.4	Conforms with the following exception: The ESBWR is designed to shut down safely without reliance on offsite or diesel-generator-derived AC power, therefore, RG 1.93 is only applicable to onsite safety-related DC power systems.
BTP 5-1	Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	Rev. 3	Mar-07		Not applicable to the ESBWR
BTP 5-2	Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures	Rev. 3	Mar-07		Not applicable to the ESBWR
BTP 5-3	Fracture Toughness Requirements	Rev. 3	Mar-07		Conforms
BTP 5-4	Design Requirements of the Residual Heat Removal System	Rev. 3	Mar-07		Not applicable to ESBWR
6.1.1	Engineered Safety Features Materials	Rev. 2	Mar-07	II.1, II.2, II.3, II.4	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
6.1.2	Protective Coating Systems (Paints) - Organic Materials	Rev. 3	Mar-07	II.1	Conforms
6.2.1	Containment Functional Design	Rev. 3	Mar-07		Conforms
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments	Rev. 3	Mar-07		Not applicable to the ESBWR
6.2.1.1.B	Ice Condenser Containments	Draft Rev. 3	Jun-96		Not applicable to the ESBWR
6.2.1.1.C	Pressure-Suppression Type BWR Containments	Rev. 7	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11	Conforms
6.2.1.2	Subcompartment Analysis	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)	Rev. 3	Mar-07	II.1, II.2, II.3	Conforms
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	Rev. 2	Mar-07		Not applicable to the ESBWR

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	Rev. 3	Mar-07		Not applicable to the ESBWR
6.2.2	Containment Heat Removal Systems	Rev. 5	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8	Conforms
6.2.3	Secondary Containment Functional Design	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms. See DCD Table 1.9-20.
6.2.4	Containment Isolation System	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11, II.12, II.13, II.14, II.15, II.16, II.17, II.18, II.19, II.20, II.21, II.22	Conforms
6.2.5	Combustible Gas Control in Containment	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9	Conforms
6.2.6	Containment Leakage Testing	Rev. 3	Mar-07		Conforms
6.2.7	Fracture Prevention of Containment Pressure Boundary	Rev. 1	Mar-07	II.1, II.2	Conforms
6.3	Emergency Core Cooling System	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.6, II.7, II.8, II.10	Conforms
				II.5, II.9	Not applicable

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
6.4	Control Room Habitability System	Rev. 3	Mar-07	II.1, II.2, II.4, II.5, II.6	Conforms
				II.3	Exception: For differential pressure testing of the control room, the periodic verification interval of every 18 months in Acceptance Criteria II.3.a through II.3.c is increased to every 24 months to accommodate the ESBWR's two year operating cycle. The frequencies for testing the CR HVAC system are defined by Technical Specifications 3.7.2 and 5.5.12 of the referenced certified design.
				II.7	Exception: SRP states that self-contained breathing apparatus for the control room personnel should be on hand. DCD Section 6.4.1.1 states that CRHA habitability requirements are satisfied without the need for individual breathing apparatus and/or special clothing.
6.5.1	ESF Atmosphere Cleanup Systems	Rev. 3	Mar-07		Conforms. Surveillances, testing, and maintenance guidelines for the CRHAVS are addressed in Technical Specifications 3.7.2, 5.5.12, and 5.5.13, Maintenance Rule requirements in Section 17.6, and procedure requirements in Section 13.5.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
6.5.2	Containment Spray as a Fission Product Cleanup System	Rev. 4	Mar-07		Not applicable. See DCD Table 1.9-20.
6.5.3	Fission Product Control Systems and Structures	Rev. 3	Mar-07	II.1, II.2 (there is no II.3)	Conforms
				II.4	Not applicable. Drywell spray function is not credited in DCD Chapter 15 dose analysis.
6.5.4	Ice Condenser as a Fission Product Cleanup System	Draft Rev. 4	Jun-96		Not applicable to the ESBWR
6.5.5	Pressure Suppression Pool as a Fission Product Cleanup System	Rev. 1	Mar-07	II.1, II.2	Conforms. Refer to DCD Table 1.9-20.
				II.3	Not applicable.
6.6	Inservice Inspection and Testing of Class 2 and 3 Components	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11	Conforms
6.7	Main Steam Isolation Valve Leakage Control System (BWR)	Draft Rev. 3	Jun-96		Not applicable
BTP 6-1	pH For Emergency Coolant Water for Pressurized Water Reactors	Initial Issuance	Mar-07		Not applicable to the ESBWR
BTP 6-2	Minimum Containment Pressure Model for PWR ECCS Performance Evaluation	Rev. 3	Mar-07		Not applicable to the ESBWR

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
BTP 6-3	Determination of Bypass Leakage Paths in Dual Containment Plants	Rev. 3	Mar-07		Conforms. Refer to DCD Table 1.9-20.
BTP 6-4	Containment Purging During Normal Plant Operations	Rev. 3	Mar-07		Conforms. Refer to TS SR 3.6.1.3.
BTP 6-5	Currently the Responsibility of Reactor Systems Piping From the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps	Rev. 3	Mar-07		Not applicable
7.0	Instrumentation and Controls - Overview of Review Process	Rev. 5	Mar-07		Conforms
Appendix 7.0-A	Review Process for Digital Instrumentation and Control Systems	Rev. 5	Mar-07		Conforms
7.1	Instrumentation and Controls - Introduction	Rev. 5	Mar-07	II.1, II.2, II.3	Conforms. Procedures addressed in Section 13.5. ITAAC addressed in COLA Part 10.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
7.1-T	Table 7-1 Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety	Rev. 5	Mar-07		Conforms
Appendix 7.1-A	Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety	Rev. 5	Mar-07	1, 2, 3, 4, 5	Conforms
Appendix 7.1-B	Guidance for Evaluation of Conformance to IEEE Std 279	Rev. 5	Mar-07		Conforms
Appendix 7.1-C	Guidance for Evaluation of Conformance to IEEE Std 603	Rev. 5	Mar-07		Conforms
Appendix 7.1-D	Guidance for Evaluation of the Application of IEEE Std 7-4.3.2	Initial Issuance	Mar-07	SRM to SECY 93-087 II.Q	Conforms
7.2	Reactor Trip System	Rev. 5	Mar-07	II.1, II.2, II.3, II.4, SRM to SECY 93-087 II.Q	Conforms. Procedures addressed in Section 13.5. Technical Specifications addressed in Chapter 16. ITAAC addressed in COLA Part 10.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
7.3	Engineered Safety Features Systems	Rev. 5	Mar-07	II.1, II.2, II.3, II.4, SRM to SECY 93-087 II.Q	Conforms. Procedures addressed in Section 13.5. Technical Specifications addressed in Chapter 16. ITAAC addressed in COLA Part 10.
7.4	Safe Shutdown Systems	Rev. 5	Mar-07	II.1, II.2, II.3	Conforms. Procedures addressed in Section 13.5. Technical Specifications addressed in Chapter 16. ITAAC addressed in COLA Part 10.
7.5	Information Systems Important to Safety	Rev. 5	Mar-07	II.1, II.2, II.3, II.4, II.5, SRM to SECY 93-087 II.Q	Conforms. Procedures addressed in Section 13.5. Technical Specifications addressed in Chapter 16. ITAAC addressed in COLA Part 10.
7.6	Interlock Systems Important to Safety	Rev. 5	Mar-07	II.1, II.2, II.3	Conforms. Procedures addressed in Section 13.5. Technical Specifications addressed in Chapter 16. ITAAC addressed in COLA Part 10.
7.7	Control Systems	Rev. 5	Mar-07	II.1, II.2, II.3, II.4, SRM to SECY 93-087 II.Q	Conforms. Procedures addressed in Section 13.5. Technical Specifications addressed in Chapter 16. ITAAC addressed in COLA Part 10.
7.8	Diverse Instrumentation and Control Systems	Rev. 5	Mar-07	II.1, II.2, II.3, II.4, SRM to SECY 93-087 II.Q	Conforms. Procedures addressed in Section 13.5. Technical Specifications addressed in Chapter 16. ITAAC addressed in COLA Part 10.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
7.9	Data Communication Systems	Rev. 5	Mar-07	II.1, II.2, II.3	Conforms. Addressed in DCD Section 7.1. Procedures addressed in Section 13.5. Technical Specifications addressed in Chapter 16. ITAAC addressed in COLA Part 10.
Appendix 7-A	General Agenda, Station Site Visits (formerly Appendix 7-B)	Rev. 5	Mar-07		Not applicable. Provides guidance to the NRC to conduct site visits.
Appendix 7-B	Acronyms, Abbreviations, and Glossary (formerly Appendix 7-C)	Rev. 5	Mar-07		Conforms
BTP 7-1	Guidance on Isolation of Low-Pressure Systems from the High-Pressure Reactor Coolant System	Rev. 5	Mar-07		Conforms
BTP 7-2	Guidance on Requirements of Motor-Operated Valves in the Emergency Core Cooling System Accumulator Lines	Rev. 5	Mar-07		Not applicable to the ESBWR
BTP 7-3	Guidance on Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service	Rev. 5	Mar-07		Not applicable to the ESBWR

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
BTP 7-4	Guidance on Design Criteria for Auxiliary Feedwater Systems	Rev. 5	Mar-07		Not applicable to the ESBWR
BTP 7-5	Guidance on Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	Rev. 5	Mar-07		Not applicable to the ESBWR
BTP 7-6	Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode	Rev. 5	Mar-07		Not applicable. ESBWR does not use recirculation pumps or active ECCS pumps.
HICB-7	Not Used				Not used
BTP 7-8	Guidance for Application of Regulatory Guide 1.22	Rev. 5	Mar-07		Conforms. Chapter 16 addresses Technical Specifications.
BTP 7-9	Guidance on Requirements for Reactor Protection System Anticipatory Trips	Rev. 5	Mar-07		Conforms
BTP 7-10	Guidance on Application of Regulatory Guide 1.97	Rev. 5	Mar-07		Conforms. Section 13.5 addresses procedures.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
BTP 7-11	Guidance on Application and Qualification of Isolation Devices	Rev. 5	Mar-07		Conforms.
BTP 7-12	Guidance on Establishing and Maintaining Instrument Setpoints	Rev. 5	Mar-07		Conforms. Section 13.5 addresses procedures.
BTP 7-13	Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors	Rev. 5	Mar-07		Not applicable. RTDs are not used in the ESBWR protection systems.
BTP 7-14	Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems	Rev. 5	Mar-07		Conforms
HCIB-15	Not Used				Not used
BTP 7-16	Withdrawn				Withdrawn
BTP 7-17	Guidance on Self-Test and Surveillance Test Provisions	Rev 5	Mar-07		Conforms. Section 13.5 addresses procedures. Chapter 16 addresses Technical Specifications.
BTP 7-18	Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems	Rev. 5	Mar-07		Conforms. Section 13.5 addresses procedures.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
BTP 7-19	Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems	Rev. 5	Mar-07		Conforms
HCIB-20	Not Used				Not used
BTP 7-21	Guidance on Digital Computer Real-Time Performance	Rev. 5	Mar-07		Conforms
8.1	Electric Power - Introduction	Rev. 3	Mar-07		Conforms
8.2	Offsite Power System	Rev. 4	Mar-07	II.4, II.5, II.6, II.8	Conforms
				II.1, II.2, II.3, II.7	Not applicable. ESBWR is a passive design and does not rely on offsite power.
8.3.1	A-C Power Systems (Onsite)	Rev. 3	Mar-07	II.1, II.2, II.3, II.4.A through II.4.H, II.4.J, II.5, II.6, II.7, II.10	Conforms
				II.4.I	Not applicable. The ESBWR diesel generators are not safety-related.
				II.8	Not applicable. The ESBWR diesel generators are not safety-related, nor is AC power needed to achieve safe shutdown.
				II.9	Conforms. Addressed in DCD Section 17.4 and in Section 17.6.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
8.3.2	D-C Power Systems (Onsite)	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.7, II.8, II.9, II.10	Conforms
				II.5, II.6	Not applicable. Addressed in DCD Sections 8.3.2.1.1 and 8.3.2.2.2.
				II.11	Not applicable. The ESBWR is designed to shutdown safely without reliance on offsite or diesel-generator-derived AC power for 72 hours, which exceeds station blackout requirements.
				II.12	Conforms. Addressed in Section 17.6.
				II.13	Conforms. Addressed in Section 17.6.
8.4	Station Blackout	Initial Issuance	Mar-07	II.1, II.2	Conforms. Addressed in DCD Section 15.5.5.
				II.3	Not applicable. Onsite Class 1E Emergency AC power sources are not required for ESBWR safe shutdown.
				II.4, II.5	Conforms. Addressed in Section 17.6.
Appendix 8-A	General Agenda, Station Site Visits	Rev. 1	Mar-07		Not applicable. Provides guidance to NRC to conduct site visits.
BTP 8-1	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	Rev. 3	Mar-07		Not applicable. The ESBWR does not have any safety-related motor-operated valves.

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SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
BTP 8-2	Use of Diesel-Generator Sets for Peaking	Rev. 3	Mar-07		Not applicable. The ESBWR can achieve safe shutdown without AC power, and the diesel-generator sets are not safety-related. Therefore, this BTP is not applicable.
BTP 8-3	Stability of Offsite Power Systems	Rev. 3	Mar-07		Conforms. Stability studies were performed to investigate the loss of off-site generation.
BTP 8-4	Application of the Single Failure Criterion to Manually Controlled Electrically Operated Valves	Rev. 3	Mar-07		Not applicable. The ESBWR does not use any manually-operated valves to mitigate an accident.
BTP 8-5	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	Rev. 3	Mar-07		Not applicable. The ESBWR is designed in accordance with ICSB 21, the predecessor to BTP 8-5, as stated in DCD Table 8.1-1 and DCD Section 8.3.2.2.2. Also, refer to DCD Table 7.1-1 for conformance to RG 1.47 and Bypass and Inoperable Status Indicator (BISI) for all safety-related systems.
BTP 8-6	Adequacy of Station Electric Distribution System Voltages	Rev. 3	Mar-07		Not applicable. The ESBWR is designed in accordance with PSB 1, the predecessor to BTP 8-6, as stated in DCD Table 8.1-1 and DCD Section 8.3.1.1.2.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
BTP 8-7	Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status	Rev. 3	Mar-07		Not applicable. The ESBWR does not use safety-related diesel generators.
9.1.1	Criticality Safety of Fresh and Spent Fuel Storage and Handling	Rev. 3	Mar-07	II.1	Conforms
9.1.2	New and Spent Fuel Storage	Rev. 4	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9	Conforms
9.1.3	Spent Fuel Pool Cooling and Cleanup System	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
				II.8	Conforms. EP-ITAAC are addressed in COLA Part 10.
9.1.4	Light Load Handling System (Related to Refueling)	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
9.1.5	Overhead Heavy Load Handling Systems	Rev. 1	Mar-07	II.1, II.2, II.3, II.4	Conforms
9.2.1	Station Service Water System	Rev. 5	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms
9.2.2	Reactor Auxiliary Cooling Water Systems	Rev. 4	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms
9.2.3	Demineralized Water Makeup System				SRP withdrawn
9.2.4	Potable and Sanitary Water Systems	Rev. 3	Mar-07	II.1.A, II.1.B, II.1.C	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
9.2.5	Ultimate Heat Sink	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms
9.2.6	Condensate Storage Facilities	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9	Conforms
9.3.1	Compressed Air System	Rev. 2	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms. Instrument Air is addressed in DCD Section 9.3.6, Service Air is addressed in DCD Section 9.3.7, and High Pressure Nitrogen Supply System is addressed in DCD Section 9.3.8.
9.3.2	Process and Post-accident Sampling Systems	Rev. 3	Mar-07	II.1, II.3, II.4	Conforms
				II.2	Exception. Technical Specifications do not require analyses. Section 9.3.2 addresses actions required to qualify process sampling for taking radioactive samples without having a specific post-accident sampling system. Analyses and frequencies of process systems are addressed in plant operating procedures.
9.3.3	Equipment and Floor Drainage System	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)	Rev. 3	Mar-07		Not applicable to the ESBWR
9.3.5	Standby Liquid Control System (BWR)	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
9.4.1	Control Room Area Ventilation System	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms. Section 9.4 was evaluated against these criteria.
9.4.2	Spent Fuel Pool Area Ventilation System	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
9.4.3	Auxiliary and Radwaste Area Ventilation System	Rev. 3	Mar-07	II.1, II.2, II.3	Conforms. Section 9.4 was evaluated against these criteria.
9.4.4	Turbine Area Ventilation System	Rev. 3	Mar-07	II.1, II.2, II.3	Conforms
9.4.5	Engineered Safety Feature Ventilation System	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms
9.5.1	Fire Protection Program	Rev. 5	Mar-07	II.1, II.2, II.4	Not applicable. See DCD Table 1.9-21.
				II.3, II.5, II.6	Conforms
				II.7	Exception: The elements of the Fire Protection Program required to be operational prior to receipt of new fuel are those elements necessary to protect buildings storing new fuel and adjacent fire areas that could affect the fuel storage area. Other required elements of the Fire Protection Program will be fully operational prior to initial fuel loading. Refer to Section 13.4.
9.5.2	Communications Systems	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11, II.12, II.13, II.14	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
9.5.3	Lighting Systems	Rev. 3	Mar-07	II.1, II.2, II.3, II.4	Conforms
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	Rev. 3	Mar-07		Not applicable to the ESBWR
9.5.5	Emergency Diesel Engine Cooling Water System	Rev. 3	Mar-07		Not applicable to the ESBWR
9.5.6	Emergency Diesel Engine Starting System	Rev. 3	Mar-07		Not applicable to the ESBWR
9.5.7	Emergency Diesel Engine Lubrication System	Rev. 3	Mar-07		Not applicable to the ESBWR
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System	Rev. 3	Mar-07		Not applicable to the ESBWR

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
10.2	Turbine Generator	Rev. 3	Mar-07	II.1.A, II.1.B	Conforms
				II.1.C	Exception—The Turbine Generator Set (TGS) has the capability to permit periodic testing of all components important to safety while the unit is at or above rated speed. In DCD Section 10.2.2.7, a list of components that may be tested with the unit at load is provided. However, some load reduction may be necessary before testing main stop and control valves, and intermediate stop and intercept valves (see DCD Section 10.2.3.7). Overspeed trip testing is performed at speed levels greater than or equal to rated speed with no electrical load. Thus, not all components are capable of being tested at rated load as required in the corresponding Acceptance Criterion.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
10.2	Turbine Generator (continued)				
				II.1.C (continued)	Load reduction for turbine valve testing is common in the existing fleet of power reactors and is considered acceptable. Testing at turbine loads below the rated load condition is considered an acceptable means of confirming that equipment relied on to prevent turbine overspeed related failures is available and capable of providing required functions. Further, component redundancies, as described in DCD Section 10.2.2.4, ensure that a single failure of any of the above valves important to safety will not disable the function of the overspeed protection system.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
10.2	Turbine Generator (continued)				
				II.2.A	Exception—Inservice inspection of main steam and reheat valves is discussed in DCD Sections 10.2.2.7 and 10.2.3.7. The first disassembly and visual inspection of all main stop valves, main control valves, intermediate stop, and intercept valves are performed within the first three refueling shutdowns. However, the interval for subsequent inspections may be extended beyond the SRP interval of 3-1/3 years to an interval consistent with applicable industry guidance, subject to the requirements of the turbine missile probability analysis. The inspection interval may not exceed the requirements or assumptions in the turbine missile probability analysis. Further, inspection intervals are only extended if there are no significant findings in the initial (baseline) inspections. Thus, with the above provisions, extending the inspection interval beyond the SRP interval is considered acceptable.
				II.2.B, II.3	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
10.2.3	Turbine Rotor Integrity	Rev. 2	Mar-07	II.1, II.2	Conforms
				II.3.A	Exception: DCD Section 10.2.3.5 states that, "Forgings are rough-machined with minimum stock allowance prior to heat treatment." This statement meets the intent of the corresponding SRP Acceptance Criterion. The exception to the Acceptance Criterion is introduced with the reference to welded rotors. The GE N3R-6F52 steam turbine selected for this site utilizes integral forgings in the rotor design and fabrication. Although other manufacturers produce welded rotors, the GE N3R-6F52 rotor is not a welded rotor design and does not utilize welding to construct the base rotor. Flaws in the forging may be repaired by welding and other means, but only after heat treatment. Thus, the intent of this Acceptance Criterion is met.
				II.3.B, II.3.C, II.3.D, II.4, II.5	Conforms
10.3	Main Steam Supply System	Rev. 4	Mar-07	II.1, II.2, II.3, II.5, II.6, II.7, II.8	Conforms
				II.4	Not applicable to the ESBWR
10.3.6	Steam and Feedwater System Materials	Rev. 3	Mar-07	II.1, II.2	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
10.4.1	Main Condensers	Rev. 3	Mar-07	II.1	Conforms
10.4.2	Main Condenser Evacuation System	Rev. 3	Mar-07	II.1	Conforms
10.4.3	Turbine Gland Sealing System	Rev. 3	Mar-07		Conforms
10.4.4	Turbine Bypass System	Rev. 3	Mar-07	II.1, II.2, II.3	Conforms
10.4.5	Circulating Water System	Rev. 3	Mar-07	II.1	Conforms
10.4.6	Condensate Cleanup System	Rev. 3	Mar-07	II.1	Conforms
				II.2	Not applicable to the ESBWR

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SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
10.4.7	Condensate and Feedwater System	Rev. 4	Mar-07	II.1, II.2.B, II.3, II.4, II.5, II.6,	Conforms
				II.2.A,	Not applicable to the ESBWR
				II.7	Exception: This SRP acceptance criterion states that guidance for acceptable FAC inspection programs "is found in (NRC) Generic Letter 89-08 and in EPRI NP-3944." EPRI document NSAC-202L, Rev. 2, supersedes EPRI NP-3944 and is therefore referenced in place of EPRI NP-3944 in DCD Section 6.6.7, for guidance regarding FAC (erosion corrosion) monitoring and related inspection programs. The more recent document, EPRI NSAC-202L, utilizes more extensive industry experience and improved inspection methods and modeling. The substitution of EPRI NSAC-202L, Rev. 2, in place of EPRI NP-3944 is therefore acceptable.
				II.8	Conforms. Addressed in DCD Sections 3.9.3, 5.2.4, and 10.4.7, and DCD Tables 1.9-22 and 1.11-1.
10.4.8	Steam Generator Blowdown System (PWR)	Rev. 3	Mar-07		Not applicable to the ESBWR

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
10.4.9	Auxiliary Feedwater System (PWR)	Rev. 3	Mar-07		Not applicable to the ESBWR
BTP 10-1	Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants	Rev. 3	Mar-07		Not applicable to the ESBWR
BTP 10-2	Design Guidelines for Avoiding Water Hammers in Steam Generators	Rev. 4	Mar-07		Not applicable to the ESBWR
11.1	Source Terms	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.6, II.7, II.8, II.9	Conforms. Addressed in DCD Section 12.2 and in Section 12.2.
				II.5	Conforms. Addressed in Sections 11.2 and 11.3.
11.2	Liquid Waste Management System	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms. Addressed in DCD Sections 11.2 and 12.2, and in Sections 11.2 and 12.2.
				II.6	Not applicable. Applies to ESP applications.
11.3	Gaseous Waste Management System	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms. Addressed in DCD Sections 11.3 and 12.2, and in Sections 11.2 and 12.2.
				II.8	Not applicable. Applies to ESP applications.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
11.4	Solid Waste Management System	Rev. 3	Mar-07	II.1, II.2, II.5, II.7, II.8, II.9, II.10, II.14	Conforms.
				II.3, II.4, II.6, II.11. II.12, II.13	Conforms (addressed in DCD Section 11.4 and in Section 11.4; for Acceptance Criterion II.13, this is also addressed in Section 11.5) with the following exception: RG 1.206, Section 13.4 includes the PCP as an operational program, and only requires a program description in the COLA and a milestone for full program implementation. The FSAR provides a description of the PCP, along with the implementation milestone. Procedures for handling waste will be developed once the PCP is implemented.

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SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	Rev. 4	Mar-07	II.1, II.2	Conforms (addressed in DCD Section 11.5.2) with the following exception: Procedural controls are based on NQA-1, rather than RG 1.33, as described in Section 13.5. Quality Assurance Program requirements are addressed in Section 17.5.
				II.3, II.4, II.5	Conforms (addressed in DCD Sections 11.5.2 and 11.5.3, and in Section 11.5) with the following exceptions: 1) RG 1.206, Section 13.4 includes the ODCM (including the SREC) and PCP as operational programs, and only requires program descriptions in the COLA and milestones for full program implementation. The FSAR provides descriptions of the PCP and ODCM along with implementation milestones. 2) Procedural controls are based on NQA-1, rather than RG 1.33, as described in Section 13.5. Quality Assurance Program requirements are addressed in Section 17.5. Conformance with NUREG-0718 is addressed in DCD Table 1.9-8.
				II.6	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
BTP 11-3	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	Rev. 3	Mar-07	B.1,B.3, B.5	Conforms
				B.2, B.4	Conforms (addressed in DCD Section 11.4 and in Section 11.4; for Acceptance Criterion II.13, this is also addressed in Section 11.5) with the following exception: RG 1.206, Section 13.4 includes the PCP as an operational program, and only requires a program description in the COLA and a milestone for full program implementation. The FSAR provides a description of the PCP, along with the implementation milestone. Procedures for handling waste will be developed once the PCP is implemented.
BTP 11-5	Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	Rev. 3	Mar-07		Conforms. Addressed in DCD Section 11.3.
BTP 11-6	Postulated Radioactive Releases Due to Liquid-containing Tank Failures	Rev. 3	Mar-07		Conforms. Addressed in DCD Section 15.3.16 and in Section 2.4.13.
12.1	Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	Rev. 3	Mar-07	II.1, II.2. II.3, II.4	Conforms. Addressed in Section 13.2, and Appendices 12AA and 12BB.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
12.2	Radiation Sources	Rev. 3	Mar-07	II.1	Not applicable. Acceptance criterion cites RG 1.3. SRP states RG 1.3 is applicable to license holders issued prior to January 10, 1997. COL applicant is not a license holder.
				II.2	Not applicable to the ESBWR
				II.3	Conforms. Addressed in DCD Sections 12.3 and 15.4 and in Section 6.4.
				II.4	Conforms. Addressed in DCD Section 12.3.
				II.5	Conforms
				II.6	Conforms. Addresses in DCD Sections 1A and 12.2.
				II.7	Conforms. Addressed in DCD Section 12.2.
12.3–12.4	Radiation Protection Design Features	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
12.5	Operational Radiation Protection Program	Rev. 3	Mar-07	II.1	Conforms with the following exceptions: 1) NUREG-0731 is not active, and is not utilized; 2) RG 8.8 specifies the use of RG 1.16. Reporting per C.1.b(2) and C.1.b(3) of RG 1.16 is no longer required.
				II.2.A, II.2.B, II.2.C, II.2.D, II.2.E.i, II.2.E.ii, II.2.E.iii, II.2.E.iv, II.2.F, II.2.G, II.2.H, II.4	Conforms
				II.2.E.v	Conforms with the following exception: NUREG-1736 states that RGs 8.20, 8.26, and 8.32 are outdated and recommends use of the methods in RG 8.9, Rev. 1. Therefore, the methods identified in RG 8.9, Rev. 1 will be used in place of those in RGs 8.20, 8.26, and 8.32.
				II.3	Conforms with the following exceptions: 1) RG 8.25 is not applicable to power stations; 2) NUREG-1736 states that RGs 8.20, 8.26, and 8.32 are outdated and recommends use of the methods in RG 8.9, Rev. 1; and 3) RP program and procedures are established, implemented, maintained, and reviewed under the QA Program described in Section 17.5.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
13.1.1	Management and Technical Support Organization	Rev. 5	Mar-07	II.1.A, B, D, II.2.A.i through II.2.A.v	Conforms
				II.1.C	Exception: The experience requirements of corporate staff are set by corporate policy and not provided in detail; however, the experience level of Dominion, as discussed in Section 13.1 and Appendix 13AA, in the area of nuclear plant development, construction, and management establishes that Dominion has the necessary capability and staff to ensure that design and construction of the facility will be performed in an acceptable manner.
				II.2.A.vi, II.2.A.vii	Conforms. Addressed in Sections 13.1 and 14.2.
				II.2.A.viii	Not applicable. Only applies to applicants whose applications were pending as of February 16, 1982.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
13.1.2–13.1.3	Operating Organization	Rev. 6	Mar-07	General 1	Exception: SRP requires operational, onsite technical support, and maintenance groups to be under the direction and supervision of a plant manager. Dominion has organized much of its technical support with direct reporting to offsite/corporate organizations and dotted line reporting to the site executive in charge of plant management. This applies to such groups as training, security, emergency preparedness, QA, licensing, and projects.
				General 2, General 3	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
13.1.2–13.1.3 *(continued)*	Operating Organization	Rev. 6	Mar-07	General 4	Not applicable. There are no requests for exemptions from the requirements of 10 CFR 50.54(m).
				II.1.A, II.1.B	Conforms with the following exception: Section 17.5 states, “The operational phase quality assurance program requirements will be established through the Company’s commitment to ANSI/ASME NQA-1-1994 as described within this QAPD. This edition of NQA-1 contains overall quality assurance requirements equivalent to those of ANSI N18.7-1976, and the Company has included within this QAPD the required administrative controls from ANSI N18.7-1976. Therefore, the Company does not commit to compliance with the requirements of ANSI N18.7-1976/ANS-3.2.”
				II.1.A.i through II.1.A.v, II.1.C, II.1.E, II.1.F, II.1.G	Conforms
				II.1.D	Not applicable
				II.1.H	Conforms. Addressed in Section 13.2.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
13.2.1	Reactor Operator Requalification Program: Reactor Operator Training	Rev. 3	Mar-07	II.1.A.i	Conforms. Addressed in Section 13.1.
				II.1.A.ii, II.1.A.iii, II.1.A.v, II.1.B, II.1.D, II.1.E	Conforms
				II.1.A.iv	Conforms. Addressed in Sections 13.1, 13.2, and 17.5.
				II.1.A.vi	Conforms. Addressed in DCD Chapter 18.
				II.1.A.vii	Exception: The COLA incorporates by reference approved industry template NEI 06-13, which does not address compliance with NUREG-1021.
				II.1.C	Exception: This item states that “formal segments of the initial licensed operator training program should be substantially complete when the pre-operational program test begins.” Appendix 13BB commits to a similar state of readiness: “Before initial fuel loading, the number of persons trained in preparation for RO and SRO licensing examinations will be sufficient to meet regulatory requirements, with allowances for examination contingencies and without the need for planned overtime.”

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
13.2.2	Non-Licensed Plant Staff Training	Rev. 3	Mar-07	II.1, II.2, II.3, II.4, II.5, II.7, II.8, II.9	Conforms.
				II.6	Exception: This item states that “formal segments of the initial training program should be substantially complete when the pre-operational test program begins.” Appendix 13BB commits to a similar state of readiness: “Before initial fuel loading, sufficient plant staff will be trained to provide for safe plant operations.”
				II.10	Conforms. Addressed in DCD Section 9.5.1.
				II.11	Conforms. Addressed in Sections 13.2 and 13.4.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
13.3	Emergency Planning	Rev. 3	Mar-07	II.1, II.2,	Conforms. Addressed in Section 13.4, COLA Part 5, and COLA Part 10.
				II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II.11, II.12, II.13, II.17, II.18, II.27, II.28, II.29, II.30	Conforms. Addressed in COLA Part 5.
				II.14	Not applicable. Allows NRC to issue a license when applicant asserts that noncompliance with offsite EP requirements is because state or local government has declined to participate in emergency planning.
				II.15, II.16, II.19, II.20, II.21	Not applicable. Only applies to ESP applications.
				II.22	Not applicable. Only applies to design certification applications.
				II.23, II.24	Conforms. Addressed in COLA Part 10.
				II.25	Conforms. Addressed in DCD Section 13.3 and COLA Part 5. The NAPS Units 1 and 2 EOF will be used for Unit 3.
				II.26	Conforms. Reviewed under SRPs 7.5 and 18.2.
				II.31	Conforms. Addressed in Section 13.4.
13.4	Operational Programs	Rev. 3	Mar-07		Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
13.5.1.1	Administrative Procedures - General	Initial Issuance	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
				II.8	Section 13.5 and DCD Section 18.9 discuss conformance with NUREG- 0711
				II.9, II.10, II.11, II.12, II.13, II.14, II.15, II.16, II.17, II.18, II.19, II.20	Conforms
13.5.2.1	Operating and Emergency Operating Procedures	Rev. 2	Mar-07	II.1	Conforms
				II.2.A, II.2.B	Conforms
				II.2.C	Section 13.5 and DCD Section 18.9 discuss conformance with NUREG- 0711
				II.2.D, II.2.E, II.2.F, II.2.G, II.2.H, II.2.I	Conforms
13.6	Physical Security	Rev. 3	Mar-07		Addressed in COLA Part 8.
13.6.1	Physical Security - Combined License Review Responsibilities	Initial Issuance	Mar-07		Addressed in COLA Part 8.
13.6.2	Physical Security - Design Certification	Initial Issuance	Mar-07		Not applicable. Applies to design certification applications.
13.6.3	Physical Security - Early Site Permit	Initial Issuance	Mar-07		Not applicable. Applies to ESP applications.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
14.2	Initial Plant Test Program - Design Certification and New License Applicants	Rev. 3	Mar-07	1A, 1B, 1C, 2A, COL/OL Applicants: 3A, 3B, 3C, 3D, 3E, 3F, 3G, 3H, 4A, 4B, 5A, 5B, 5C, 5D, 6A, 6B, 6C	Conforms with the following exception: Refer to Table 1.9-202 for exceptions to RG 1.68.
				DC Applicants: 3A, 3B, 3C, 3D, 4A, 6A, 6B, 6C	Not applicable. Applies to DC applicants.
14.2.1	Generic Guidelines for Extended Power Uprate Testing Programs	Initial Issuance	Aug-06		Not applicable. Applies to power uprates.
14.3	Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1, II.2	Conforms
14.3.1	[Reserved]	[Reserved]	Mar-07		Not used
14.3.2	Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II.9, II.10, II. 11	Conforms
14.3.3	Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1, II.2.A, II.2.B, II.2.C, II.2.D, II.2.E	Conforms
14.3.4	Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
14.3.5	Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms
14.3.6	Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	Class 1E Equipment: II.1, II.2, II.3, II.4, II.5 Other Electrical Equipment Important to Safety: II.1, II.2, II.3, II.4, II.5	Conforms
14.3.7	Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7, II.8, II. 9	Conforms
14.3.8	Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1, II.2, II.3	Conforms
14.3.9	Human Factors Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1, II.2, II.3, II.4, II.5, II.6	Conforms
14.3.10	Emergency Planning - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1, II.2	Conforms
14.3.11	Containment Systems - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1, II.2, II.3, II.4, II.5	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
14.3.12	Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria	Initial Issuance	Mar-07	II.1	Conforms
15	Introduction - Transient and Accident Analyses	Rev. 3	Mar-07	I.1, I.2, 1.3, I.4, I.5, I.6	Conforms
15.0.1	Radiological Consequence Analyses Using Alternative Source Terms	Rev. 0	Jul-00	V	Conforms
15.0.2	Review of Transient and Accident Analysis Method	Rev. 0	Dec-05	II.1, II.2, II.3, II.4, II.5, II.6	Conforms
15.0.3	Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors	Initial Issuance	Mar-07		Not applicable to the ESBWR
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	Mar-07	II.1, II.2, II.3, II.4, II.5, 1, 2, 3, 4	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)	Rev. 3	Mar-07		Not applicable to the ESBWR
15.1.5.A	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR				Not applicable to the ESBWR
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	Mar-07	1A, 1B, 1C, 1D, 2A, 2B, 2D, 2E, 2F, 3A, 3B, 3C, 3D	Conforms
				2C	Not applicable. This is not an event of moderate frequency.
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	Rev. 2	Mar-07	II.1, II.2, II.4, II.5, II.5B, II.5C, II.5D	Conforms
				II.3	Not applicable. This is not an event of moderate frequency.
				II.5A	Not applicable. There are no RCS loops in the ESBWR.
15.2.7	Loss of Normal Feedwater Flow	Rev. 2	Mar-07	1A, 1B, 1C, 1D, 2A, 2B, 2D, 2E, 2F, 3A, 3B, 3C, 3D	Conforms
				2C	Not applicable. This is not an event of moderate frequency.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	Rev. 2	Mar-07		Not applicable to the ESBWR
15.3.1–15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	Rev. 2	Mar-07		Not applicable to the ESBWR
15.3.3–15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	Rev. 3	Mar-07		Not applicable to the ESBWR
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	Rev. 3	Mar-07	1A, 1C	Conforms
				1B	Not applicable to the ESBWR
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	Rev. 3	Mar-07	1A, 1C	Conforms
				1B	Not applicable to the ESBWR
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	Rev. 3	Mar-07	1, 2, 3	Conforms

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
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	Mar-07	1A, 1B, 1D, 1E, 1F, 1, 2, 3, 4	Conforms
				1C	Not applicable. This is not an event of moderate frequency.
15.4.6	Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)	Rev. 2	Mar-07		Not applicable to the ESBWR
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Rev. 2	Mar-07	1, 2	Conforms
15.4.8	Spectrum of Rod Ejection Accidents (PWR)	Rev. 3	Mar-07		Not applicable to the ESBWR
15.4.8.A	Radiological Consequences of a Control Rod Ejection Accident (PWR)				Not applicable to the ESBWR
15.4.9	Spectrum of Rod Drop Accidents (BWR)	Rev. 3	Mar-07	1, 2, 3	Conforms. Postulated events are not applicable to the ESBWR.
15.4.9.A	Radiological Consequences of Control Rod Drop Accident (BWR)	Rev 2	July 81		Conforms. Postulated control rod drop events are not applicable to the ESBWR.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
15.5.1–15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Rev. 2	Mar-07	1, 2, 3	Conforms
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	Rev. 2	Mar-07	1, 2, 3, A, B, C, D	Conforms
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	Rev. 2	Jul-81	II.1, II.2	Conforms
15.6.3	Radiological Consequences of Steam Generator Tube Failure				Not applicable to the ESBWR
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	Rev. 2	Jul-81	II.1, II.2, II.3	Conforms
				II.4	Conforms. Addressed in TS 3.4.3.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
15.6.5	Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Rev. 3	Mar-07	II.1A, II.1B, II.1C, II.1D, II.1.E, II.2, II.3	Conforms.
15.6.5.A	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution	Rev 1	July 81		Not Applicable. Reference DCD Table 1.9-20.
15.6.5.B	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment	Rev 1	July 81		Not Applicable. Reference DCD Table 1.9-20.
15.6.5.D	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Main Steam Isolation Valve Leakage Control System (BWR)	Rev 1	July 81		Not Applicable. Reference DCD Table 1.9-20.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures			1, 2	Conforms
15.7.4	Radiological Consequences of Fuel Handling Accidents	Rev. 2	Jul-81	II.1, II.2, II.3, II.4, II.5	Conforms. Radiological assumptions superseded by SRP 15.0.1.
15.7.5	Spent Fuel Cask Drop Accidents	Rev. 2	July 81	II.1, II.2, II.3, II.4, II.5	Conforms. Because a spent fuel cask drop exceeding 9.2 m (30 ft) is not postulated (DCD Section 15.4.10.1), per SRP 15.7.5 a design basis radiological analysis is not required. Therefore, the acceptance criteria do not apply even though the SRP does.
15.8	Anticipated Transients Without Scram	Rev. 2	Mar-07	1A	Not applicable. ESBWR does not have recirculation pumps.
				1B, 1C, 1D, 1E	Conforms
				1F	Conforms
15.9	Boiling Water Reactor Stability	Initial Issuance	Mar-07	1, 2, 3, 4A, 4B, 5, 6, 7, 9A, 9B, 9C, 10, 11	Conforms
				8, 9D	Conforms
16	Technical Specifications	Rev. 2	Mar-07		Conforms
16.1	Risk-informed Decision Making: Technical Specifications	Rev. 1	Mar-07		Not applicable

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
17.1	Quality Assurance During the Design and Construction Phases	Rev. 2	Jul-81		Not applicable. RG 1.206 refers the COL applicant to Section 17.5 for the format and content of a QA Program for design and construction of new plants.
17.2	Quality Assurance During the Operations Phase	Rev. 2	Jul-81		Not applicable. RG 1.206 refers the COL applicant to Section 17.5 for the format and content of a QA Program for design and construction of new plants.
17.3	Quality Assurance Program Description	Rev. 0	Aug-90		Not applicable. RG 1.206 refers the COL applicant to Section 17.5 for the format and content of a QA Program for design and construction of new plants.
17.4	Reliability Assurance Program (RAP)	Initial Issuance	Mar-07	II.B.1, II.B.2, II.B.3, II.B.4, II.B.5, II.B.6, II.B.7, II.B.8, II.B.9	Conforms. Addressed in DCD Section 17.4 and in Section 17.6.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
17.5	Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	Initial Issuance	Mar-07	II.A, II.B.1, II.B.2, II.B.3, II.B.4, II.B.5, II.B.6, II.B.7, II.C, II.D, II.E, II.F.1, II.F.2, II.F.3, II.F.4, II.F.5, II.F.6, II.F.7, II.F.9, II.F.12, II.G, II.H, II.I, II.J, II.K, II.L.1, II.L.2, II.L.3, II.L.4, II.L.5, II.L.6, II.L.7, II.M.1, II.M.2, II.M.3, II.M.4, II.M.5, II.N, II.O, II.P, II.Q, II.R.1, II.R.2, II.R.3.a, II.R.3.c, II.R.4, II.R.5, II.R.6, II.R.7, II.R.8, II.R.9, II.R.10, II.R.11, II.R.12, II.S, II.T, II.U.1.a, II.U.1.b, II.U.1.c, II.U.1.d, II.U.2.a, II.U.2.b, II.U.2.c, II.U.2.d, II.U.2.e, II.U.2.f, II.U.2.g, II.U.2.h, II.U.2.i, II.U.2.j, II.U.2.l, II.V	DOM-QA-1: Conforms
				II.B.8	DOM-QA-1: Alternative language addresses the grace period (previously approved by NRC).
				II.B.9, II.F.8, II.F.10, II.F.11, II.M.6, II.M.7, II.M.8, II.R.3.b, II.W	DOM-QA-1: Not applicable. DOM-QA-1 is not used during the operational phase.
				II.L.8	DOM-QA-1: Not applicable. This process for qualification of commercial-grade calibration services is not used.
				II.U.1.e	DOM-QA-1: Not a commitment in DOM-QA-1. Included in implementing procedure.
				II.U.2.k	DOM-QA-1: Not applicable. On-line records not used.

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Table 1.9-201 Conformance with Standard Review Plan

SRP Section	Title	Rev	Date	Specific Acceptance Criteria	Evaluation
17.5 *(continued)*	Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	Initial Issuance	Mar-07	II.A, II.B, II.C, II.D., II.E, II.F, II.G, II.H, II.I, II.J, II.K, II.L, II.M, II.N, II.O, II.P, II.Q, II.R, II.S, II.T, II.U, II.V, II.W Option 1	Dominion QAPD (Appendix 17AA): Conforms
				II.W Option II	Dominion QAPD: Not applicable for North Anna. Option I chosen
17.6	Maintenance Rule	Initial Issuance	Mar-07	II.1, II.2	Conforms
18	Human Factors Engineering	Rev. 2	Mar-07	II.A	Conforms
				II.B, II.C	Not applicable. These acceptance criteria apply to changes to existing plants.
19.0	Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors	Rev. 2	Jun-07	II.1, II.2, II.3, II.4, II.5, II.6, II.7	Conforms
				II.8, II.9	Not applicable. Only applies to Westinghouse AP 600 design.
19.1	Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	Rev. 2	Jun-07		Not applicable. There are no plans for risk-informed activities.
19.2	Review of Risk Information Used to Support Permanent Plant Specific Changes to the Licensing Basis: General Guidelines	Rev. 0	Jun-07		Not applicable. There are no plans for risk-informed applications.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	Rev. 0	Nov-70	General	Not applicable
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors	Rev. 2	Jun-74	General	Not applicable. RG 1.183 is used.
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors	Rev. 2	Jun-74	General	Not applicable
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors	Rev. 0	Mar-71	General	Not applicable. RG 1.183 is used.
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	Rev. 0	Mar-71	General	Not applicable
1.7	Control of Combustible Gas Concentrations in Containment	Rev. 3	Mar-07	General	Conforms

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.8	Qualification and Training of Personnel for Nuclear Power Plants	Rev. 3	May-00	C.1	Conforms.
				C.2	Conforms, with the following exceptions: (1) instead of NQA-1-1983 or NQA-1-1989, NQA-1-1994 is utilized as specified in the QAPD; (2) experience requirements cannot be met prior to operations as described in Appendix 13BB.
1.9	Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants	Rev. 4	Mar-07	General	Not applicable
1.11	Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11) Supplement to Safety Guide 11, Backfitting Considerations	Rev. 0	Feb-72	C.1, C.2, E	Conforms
1.12	Nuclear Power Plant Instrumentation for Earthquakes	Rev. 2	Mar-97	C.1, C.4 – C.7	Conforms
				C.3, C.8	Conforms. The seismic monitoring program, including the necessary test and operating procedures, will be implemented prior to receipt of fuel on site.
1.13	Spent Fuel Storage Facility Design Basis	Rev. 2	Mar-07	General	Conforms

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.14	Reactor Coolant Pump Flywheel Integrity	Rev. 1	Aug-75	General	Not applicable
1.16	Reporting of Operating Information–Appendix A Technical Specifications	Rev. 4	Aug-75	General	Conforms with the following exceptions: Reporting per C.1.b(2) and C.1.b(3) is no longer required
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	Rev. 3	Mar-07	C.1	Conforms.
				C.2	Conforms
				C.3	Conforms
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	Jun-74	General	Conforms. Sections 11.4.2.3 (NEI 07-10) and 11.5.4.5 (NEI 07-09) provide descriptions of the PCP and ODCM, respectively. Implementation milestones are provided in Section 13.4.
1.22	Periodic Testing of Protection System Actuation Functions	Rev. 0	Feb-72	General	Conforms. Operational program implementation is described in Section 13.4.
1.23	Meteorological Monitoring Programs for Nuclear Power Plants	Rev. 1	Mar-07	General	Exception. Conform to Proposed Revision 1 to RG 1.23. See SSAR Section 1.8.2.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	Rev. 0	Mar-72	All	Not applicable
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors	Rev. 0	Mar-72	General	Not applicable. RG 1.183 is used.
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	Rev. 4	Mar-07	All	Exception: The requirements for quality group classifications and standards are defined by the DCD which implements Rev. 3. Refer to DCD Tables 1.9-21, 1.9-21a, 1.9-21b.
		Rev. 3	Feb-76	All	Conforms. Refer to DCD Tables 1.9-21, 1.9-21a, 1.9-21b.
1.27	Ultimate Heat Sink for Nuclear Power Plants	Rev. 2	Jan-76	General	The UHS is within the scope of the referenced certified design and is addressed in DCD Section 9.2.5.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.28	Quality Assurance Program Requirements (Design and Construction)	Rev. 3	Aug-85	General	Exception: The QAPD identified in Section 17.5 addresses a QA program based on the newer NQA-1-1994, as provided for in SRP 17.5.
1.29	Seismic Design Classification	Rev. 4	Mar-07	General	Exception: The requirements for seismic design classification are defined by the DCD which implements Rev. 3. Refer to DCD Tables 1.9-21, 1.9-21a, 1.9-21b.
		Rev. 3	Sep-78	All	Conforms. Refer to in DCD Tables 1.9-21, 1.9-21a, 1.9-21b.
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	Rev. 0	Aug-72	General	Exception: The QAPD identified in Section 17.5 addresses a QA program based on a newer NQA-1-1994, as discussed in SRP 17.5.
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	Rev. 3	Apr-78	General	Conforms. Operational program implementation is described in Section 13.4.
1.32	Criteria for Power Systems for Nuclear Power Plants	Rev. 3	Mar-04	General	Conforms.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.33	Quality Assurance Program Requirements (Operation)	Rev. 2	Feb-78	General	Exception. The QAPD topical report identified in Section 17.5 follows NQA-1 rather than the older standards referenced in RG 1.33.
1.34	Control of Electroslag Weld Properties	Rev. 0	Dec-72	General	Conforms. Operational program implementation is described in Section 13.4.
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments	Rev. 3	Jul-90	General	Not applicable
1.35.1	Determining Prestressing for Inspection of Prestressed Concrete Containments	Rev. 0	Jul-90	General	Not applicable
1.36	Nonmetalic Thermal Insulation for Austenitic Stainless Steel	Rev. 0	Feb-73	General	Conforms. Operational program implementation is described in Section 13.4.
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	Rev. 1	Mar-07	General	Conforms

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	Rev. 2	May-77	General	Exception. Section 17.5 identifies equivalent quality assurance standards.
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	Rev. 2	Sep-77	General	Exception. Section 17.5 identifies equivalent quality assurance standards.
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants	Rev. 0	Mar-73	General	Conforms
1.41	Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments	Rev. 0	Mar-73	General	Conforms with the following exception: There are no safety-related DGs for ESBWR.
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	Rev. 0	May-73	General	Conforms
1.44	Control of the Use of Sensitized Stainless Steel	Rev. 0	May-73	General	Conforms. Operational program implementation is described in Section 13.4.
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	Rev. 0	May-73	General	Conforms. Operational program implementation is described in Section 13.4.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	Rev. 0	May-73	General	Conforms. Operational program implementation is described in Section 13.4.
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	Rev. 0	May-73	General	Conforms. Operational program implementation is described in Section 13.4.
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	Jun-01	General	Conforms
1.53	Application of the Single-Failure Criterion to Safety Systems	Rev. 2	Nov-03	General	Conforms
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants	Rev. 1	Jul-00	General	Conforms
1.56	Maintenance of Water Purity in Boiling Water Reactors	Rev. 1	Jul-78	General	Conforms.
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	Rev. 1	Mar-07	General	Conforms

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.59	Design Basis Floods for Nuclear Power Plants (Errata Published 7/30/80)	Rev. 2	Aug-77	General	Conforms
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	Rev. 1	Dec-73	General	Conforms
1.61	Damping Values for Seismic Design of Nuclear Power Plants	Rev. 1	Mar-07	General	Conforms
1.62	Manual Initiation of Protective Actions	Rev. 0	Oct-73	General	Conforms
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants	Rev. 3	Feb-87	General	Conforms
1.65	Materials and Inspections for Reactor Vessel Closure Studs	Rev. 0	Oct-73	General	Conforms
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants	Rev. 2	Aug-78	General	Conforms with the following exception: Equipment listed in Appendix A, Items 1.k(2) and 1.k(3) not included in the initial test program.
1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants	Rev. 1	Jan-77	General	Conforms

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	Rev. 1	Jul-78	General	Conforms
1.68.3	Preoperational Testing of Instrument and Control Air Systems	Rev. 0	Apr-82	General	Conforms
1.69	Concrete Radiation Shields for Nuclear Power Plants	Rev. 0	Dec-73	General	Conforms
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants LWR Edition	Rev. 3	Nov-78	—	Not applicable. RG 1.206 is used. Table 1.9-203.
1.71	Welder Qualification for Areas of Limited Accessibility	Rev. 1	Mar-07	General	Conforms. Operational program implementation is described in Section 13.4.
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin	Rev. 2	Nov-78	General	Not applicable
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	Rev. 0	Jan-74	General	Conforms

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.75	Criteria for Independence of Electrical Safety Systems	Rev. 3	Feb-05	General	Conforms
1.76	Design Basis Tornado and Tornado Missiles for Nuclear Power Plants	Rev. 1	Mar-07	General	Conforms
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	Rev. 0	May-74	General	Not applicable
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	Rev. 1	Dec-01	General	Conforms
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	Rev. 1	Sep-75	General	Not applicable
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	Rev. 1	Jan-75	General	Not applicable
1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident	Rev. 3	Nov-03	General	Conforms

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes	Rev. 1	Jul-75	General	Not applicable
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III	Rev. 33	Aug-05	General	Conforms
1.86	Termination of Operating Licenses for Nuclear Reactors	Rev. 0	Jun-74	General	This RG is outside the scope of the FSAR.
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596)	Rev. 1	Jun-75	General	Not applicable
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	Rev. 1	Jun-84	General	Conforms. Source terms from RG 1.183 used.
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	Rev. 1	Aug-77	General	Not applicable

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants	Rev. 1	Feb-78	General	Conforms
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	Rev. 2	Jul-06	General	Conforms
1.93	Availability of Electric Power Sources	Rev. 0	Dec-74	General	Conforms with the following exception: The ESBWR is designed to shut down safely without reliance on offsite or diesel-generator-derived AC power, therefore, the regulatory guide is only applicable to onsite safety-related DC power systems.
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	Apr-76	General	Exception. Section 17.5 identifies equivalent QA standards in NQA-1, Subpart 2.5.
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	Rev. 1	Jun-76	General	Not applicable

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants	Rev. 4	Jun-06	General	Conforms. Operational program implementation is described in Section 13.4.
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor	Rev. 0	Mar-76	General	Not applicable. Superseded by BTP 11-5.
1.99	Radiation Embrittlement of Reactor Vessel Materials	Rev. 2	May-88	General	Conforms. Operational program implementation is described in Section 13.4.
1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants	Rev. 2	Jun-88	General	Conforms
1.101	Emergency Response Planning and Preparedness for Nuclear Power Reactors	Rev. 5	Jun-05	General	Not applicable
1.102	Flood Protection for Nuclear Power Plants	Rev. 1	Sep-76	General	Conforms
1.105	Setpoints For Safety-Related Instrumentation	Rev. 3	Dec-99	General	Conforms. Operational program implementation is described in Section 13.4.

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RG Number	Title	Revision	Date	RG Position	Evaluation
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	Rev. 1	Feb-77	General	Not applicable
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	Rev. 1	Feb-77	General	Not applicable
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	Oct-77	General	Conforms
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors	Rev. 0	Mar-76	General	Conforms
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	Rev. 1	Jul-77	General	Conforms

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors	Rev. 1	Mar-07	General	Conforms except the suggested breakdown identified in Appendix A to the RG is not used because it is not consistent with the DCD presentation of information.
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	Rev. 1	Apr-77	General	Conforms
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit	Rev. 2	May-89	General	Conforms
1.115	Protection Against Low-Trajectory Turbine Missiles	Rev. 1	Jul-77	General	Conforms
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	Rev. 0	May-77	General	Exception: Section 17.5 identifies equivalent QA standards in NQA-1, Subpart 2.8.
1.117	Tornado Design Classification	Rev. 1	Apr-78	General	Conforms
1.118	Periodic Testing of Electric Power and Protection Systems	Rev. 3	Apr-95	General	Conforms. Operational program implementation is described in Section 13.4.

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RG Number	Title	Revision	Date	RG Position	Evaluation
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes	Rev. 0	Aug-76	General	Not applicable
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	Rev. 1	Feb-78	General	Conforms
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports	Rev. 2	Feb-07	General	Conforms
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	Rev. 1	Oct-78	General	Conforms
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	Rev. 1	Mar-78	General	Conforms
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	Rev. 1	Mar-78	General	Conforms
1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	Rev. 2	Feb-07	General	Not Applicable. IEEE 484 does not apply to ESBWR VRLA batteries, therefore, RG 1.128 is not applicable. IEEE 1187 applies to VRLA batteries.

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RG Number	Title	Revision	Date	RG Position	Evaluation
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	Rev. 2	Feb-07	General	Not Applicable. IEEE 450 does not apply to ESBWR VRLA batteries, therefore, RG 1.129 is not applicable. IEEE 1188 applies to VRLA batteries.
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Supports	Rev. 2	Mar-07	General	Conforms
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants	Rev. 0	Aug-77	General	Conforms

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RG Number	Title	Revision	Date	RG Position	Evaluation
1.132	Site Investigations for Foundations of Nuclear Power Plants	Rev. 2	Oct-03	C.1, C.2, C.3, C.4.1 – C.4.2, C.4.4, C.4.5, C.5 – C.7	Conforms
				C.4.3	Conforms with the following exceptions: The RG identifies that at least one continuously sampled boring should be used for each safety-related structure. For the Unit 3 investigation, the rock was continuously cored. Because all of the soil above the rock will be removed under major structures, continuous sampling was not performed in the soil. (Continuous sampling to 15 ft depth, and the CPTs in soil provides a continuous record.) The RG identifies that boreholes with depths greater than about 100 ft should be surveyed for deviation.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.132 *(cont'd)*	Site Investigations for Foundations of Nuclear Power Plan	Rev. 2	Oct-03	C.4.3 *(cont'd)*	*(continued)* Deviation surveys were made in the three deepest boreholes in conjunction with the down-hole geophysical testing, but not in all holes deeper than 100 ft depth, since such deviation surveys serve no useful purpose. The RG identifies that color photographs of all cores should be taken soon after removal from the borehole to document the condition of the soils at the time of drilling. Color photos were taken of the rock cores but not the soil samples. The undisturbed soil samples are sealed in steel tubes. The disturbed soil samples have lost their structure and thus a photo serves little useful purpose.
1.133	Loose-Part Detection Program for the Primary System of Light Water Cooled Reactors	Rev. 1	May-81	General	Not applicable

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.134	Medical Evaluation of Licensed Personnel for Nuclear Power Plants	Rev. 3	Mar-98	General	Conforms. Although RG 1.134 is not specifically identified in the FSAR, equivalent requirements for medical evaluations for licensed personnel are embedded in policies and procedures of operations and training departments.
1.135	Normal Water Level and Discharge at Nuclear Power Plants	Rev. 0	Sep-77	General	Not applicable. Water levels and discharges in Lake Anna were evaluated in the SSAR and ESP-ER.
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments	Rev. 3	Mar-07	General	Conforms
1.137	Fuel-Oil Systems for Standby Diesel Generators	Rev. 1	Oct-79	General	Not applicable
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants	Rev. 2	Dec-03	General	Conforms
1.139	Guidance for Residual Heat Removal	Rev. 0	May-78	General	Conforms

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RG Number	Title	Revision	Date	RG Position	Evaluation
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	Jun-01	General	Conforms. Operational program implementation is described in Section 13.4.
1.141	Containment Isolation Provisions for Fluid Systems	Rev. 0	Apr-78	General	Conforms
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)	Rev. 2	Nov-01	General	Conforms
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	Rev. 2	Nov-01	General	Conforms. Operational program implementation is described in Section 13.4.
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	Rev. 1	Nov-82	General	Conforms
1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1	Rev. 14	Aug-05	General	Conforms. Operational program implementation is described in Section 13.4.

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RG Number	Title	Revision	Date	RG Position	Evaluation
1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants	Rev. 0	Mar-81	General	Conforms
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations	Rev. 3	Oct-01	General	Conforms
1.150	Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations	Rev. 1	Feb-83	General	Conforms. Operational program implementation is described in Section 13.4.
1.151	Instrument Sensing Lines	Rev. 0	Jul-83	General	Conforms. Operational program implementation is described in Section 13.4.
1.152	Criteria for Use of Computers in Safety Systems of Nuclear Power Plants	Rev. 2	Jan-06	General	Conforms. Operational program implementation is described in Section 13.4.
1.153	Criteria for Safety Systems	Rev. 1	Jun-96	General	Conforms
1.154	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors	Rev. 0	Jan-87	General	Not applicable

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RG Number	Title	Revision	Date	RG Position	Evaluation
1.155	Station Blackout	Rev. 0	Aug-88	General	Conforms, except no emergency AC power is required for the ESBWR. Only the coping analysis is applicable. Operational program implementation is described in Section 13.4.
1.156	Environmental Qualification of Connection Assemblies for Nuclear Power Plants	Rev. 0	Nov-87	General	Conforms
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance	Rev. 0	May-89	General	Conforms
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants	Rev. 0	Feb-89	General	Conforms
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors	Rev. 1	Oct-03	General	Conforms. The amount of funds for decommissioning and the method of financial assurance is described in COLA Part 1.
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	Rev. 2	Mar-97	General	Conforms. Operational program implementation is described in Section 13.4.
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb.	Rev. 0	Jun-95	General	Not applicable.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels	Rev. 0	Feb-96	General	This RG is outside the scope of the FSAR.
1.163	Performance-Based Containment Leak-Test Program	Rev. 0	Sep-95	General	Conforms
1.165	Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion	Rev. 0	Mar-97	General	Conforms. See also SSAR Section 1.8.2.
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions	Rev. 0	Mar-97	General	Conforms. The seismic monitoring program, including the necessary test and operating procedures, will be implemented prior to receipt of fuel on site.
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event	Rev. 0	Mar-97	General	Not applicable.
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 1	Feb-04	General	Conforms. Procedures addressed in Section 13.5. ITAAC addressed in COLA Part 10.
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 0	Sep-87	General	Conforms. Procedures addressed in Section 13.5. ITAAC addressed in COLA Part 10.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.170	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 0	Sep-97	General	Conforms. Procedures addressed in Section 13.5. ITAAC addressed in COLA Part 10.
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 0	Sep-97	General	Conforms. Procedures addressed in Section 13.5. ITAAC addressed in COLA Part 10.
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 0	Sep-97	General	Conforms. Procedures addressed in Section 13.5. ITAAC addressed in COLA Part 10.
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Rev. 0	Sep-97	General	Conforms. Procedures addressed in Section 13.5. ITAAC addressed in COLA Part 10.
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	Rev. 1	Nov-02	General	Not applicable. The approach described in this RG is not being used.
1.175	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing	Rev. 0	Aug-98	General	Not applicable. Risk informed inservice testing is not being used.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.176	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance	Rev. 0	Aug-98	General	Not applicable. A risk-based graded QA program is not being used.
1.177	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications	Rev. 0	Aug-98	General	Not applicable. Risk informed Technical Specifications are not being used.
1.178	An Approach For Plant-Specific Risk-informed Decisionmaking Inservice Inspection of Piping	Rev. 0	Sep-98	General	Not applicable. Risk informed inservice inspection is not being used.
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors	Rev. 0	Jan-99	General	This RG is outside the scope of the FSAR.
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	Rev. 1	Oct-03	General	Conforms. Operational program implementation is described in Section 13.4.
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)	Rev. 0	Sep-99	General	Conforms
1.182	Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants	Rev. 0	May-00	General	Conforms

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	Rev. 0	Jul-00	General	Conforms
1.184	Decommissioning of Nuclear Power Reactors	Rev. 0	Jul-00	General	Not applicable. The RG provides guidance on how to conduct decommissioning activities.
1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report	Rev. 0	Jul-00	General	This RG is outside the scope of the FSAR.
1.186	Guidance and Examples for Identifying 10 CFR 50.2 Design Bases	Rev. 0	Oct-00	General	This RG is outside the scope of the FSAR.
1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments	Rev. 0	Nov-00	General	Conforms.
1.188	Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses	Rev. 1	Sep-05	General	This RG is outside the scope of the FSAR.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.189	Fire Protection for Nuclear Power Plants	Rev. 1	Mar-07	General	Conforms with the following exception. Section C.1.1.c of the RG states that during construction, on sites with an operating unit, the superintendent of the operating plant should have overall responsibility for fire protection. However, due to physical and administrative separation of Unit 3 from the operating units, the on-site executive in charge of construction will have overall responsibility for Unit 3 fire protection during construction.
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence	Rev. 0	Mar-01	General	Conforms. The reactor vessel material surveillance program is described in Section 5.3.1.8. Implementation of the program is described in Section 13.4.
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown	Rev. 0	May-01	General	This RG is outside the scope of the FSAR.
1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code	Rev. 0	Jun-03	General	Conforms. Operational program implementation is described in Section 13.4.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.193	ASME Code Cases Not Approved for Use	Rev. 1	Aug-05	General	Conforms
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants	Rev. 0	Jun-03	General	Conforms
1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors	Rev. 0	May-03	General	Not applicable. RG 1.183 is used.
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors	Rev. 1	Jan-07	General	Conforms
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Plant Reactors	Rev. 0	May-03	General	Conforms
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction At Nuclear Power Plant Sites	Rev. 0	Nov-03	General	Conforms
1.199	Anchoring Components and Structural Supports in Concrete	Rev. 0	Nov-03	General	Conforms

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	Rev. 1	Jan-07	General	Not applicable
1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance	Rev. 1	May-06	General	Not applicable
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors	Rev. 0	Feb-05	General	Not applicable. The RG provides guidance for submitting decommissioning cost estimates to NRC prior to license termination.
1.203	Transient and Accident Analysis Methods	Rev. 0	Dec-05	General	Conforms
1.204	Guidelines for Lightning Protection of Nuclear Power Plants	Rev. 0	Nov-05	General	Conforms. Operational program implementation is described in Section 13.4.
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	Rev. 0	May-06	General	Not applicable. Risk-informed, performance-based fire protection is not used.
1.206	Combined License Applications for Nuclear Power Plants (LWR Edition)	Rev. 0	Jun-07	General	See Table 1.9-203.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
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	Mar-07	General	Conforms
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion	Rev. 0	Mar-07	All	Not applicable. The RG 1.208 performance-based approach to define the SSE ground motion is not used. See Section 2.5.2 and SSAR Section 2.5.2.
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	Rev. 0	Mar-07	General	Conforms. Operational program implementation is described in Section 13.4.
4.7	General Site Suitability Criteria for Nuclear Power Stations	Rev. 2	Apr-98	General	Conforms. See SSAR Section 1.8.2.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
4.15	Quality Assurance for Radiological Monitoring Programs (Inception Through Normal Operations to License Termination) – Effluent Streams and the Environment	Rev. 1	Feb-79	General	Conforms. Section 11.5.4.5 (NEI 07-09) provides a description of the ODCM. The implementation milestone is provided in Section 13.4. Justification for referring to RG 4.15 Rev 1 instead of Rev 2 Dominion will extend the existing North Anna Units 1 and 2 program for quality assurance of radiological effluent and environmental monitoring, that is based on Regulatory Guide 4.15, Revision 1, to apply to North Anna Unit 3. Regulatory Guide 4.15, Revision 1 is a proven methodology for quality assurance of radiological effluent and environmental monitoring programs that is acceptable to the NRC staff as a method for demonstrating compliance with applicable requirements of 10 CFR Parts 20, 50, 52, 61, and 72. Use of Revision 2 of Regulatory (continued)

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
4.15 *(cont'd)*	Quality Assurance for Radiological Monitoring Programs (Inception Through Normal Operations to License Termination) – Effluent Streams and the Environment	Rev. 1	Feb-79	General	Guide 4.15 would necessitate conducting two separate programs involving the use of common staff, facilities, and equipment, which would create an undue burden and may lead to increased probability for human error. Therefore, Dominion commits to use RG 4.15, Revision 1 methodology for North Anna Unit 3 for optimal consistency, efficiency, and practicality.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
5.44	Perimeter Intrusion Alarm Systems	Rev. 3	Oct-97	C.1.1(2), C.1.1(3), C.1.1.1 – C.1.1.5, C.1.2 – C.1.7.1, C.1.8, C.2.1, C.2.2, C.2.4, C.2.8, C.3.1	Conforms
				C.1.1(1)	Exception. The RG states that one individual should be able to assess a zone of 100 m or 328 ft from the end of that zone. There is one zone that is longer than the recommended 100 m; however, this zone has two individuals tasked with the coverage over this zone and there is CCTV coverage over a portion of that zone as an added enhancement.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
5.44 *(cont'd)*	Perimeter Intrusion Alarm Systems	Rev. 3	Oct-97	C.1.7.2	Exception. North Anna's BREs are positioned so that the officers can observe multiple zones in two directions in what could be considered a "V" shape. This is not consistent with the RG guidance "the guard observing in one direction," but it is evaluated as being effective considering the detection systems and BRE configuration in relationship to the isolation zones.
				C.2.3, C.2.5 – C.2.7	Not applicable. These types of detection equipment are not used.
				C.3.2	Not applicable. This testing option is not used.
5.62	Reporting of Safeguards Events	Rev. 1	Nov-87	General	Conforms
5.66	Access Authorization Program for Nuclear Power Plants	Rev. 0	Jun-91	General	Not applicable. NEI 03-01, Revision 1, April 2004 is used.
8.1	Radiation Symbol	Rev. 0	Feb-73	General	Conforms. The facility utilizes standard radiation symbols.
8.2	Guide for Administrative Practices in Radiation Monitoring	Rev. 0	Feb-73	General	Conforms. Operational program implementation is described in Section 13.4.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters	Rev. 0	Feb-73	General	Conforms. Operational program implementation is described in Section 13.4.
8.5	Criticality and Other Interior Evacuation Signals	Rev. 1	Mar-81	General	Conforms. Operational program implementation is described in Section 13.4.
8.6	Standard Test Procedure for Geiger-Muller Counters	Rev. 0	May-73	General	Conforms. Operational program implementation is described in Section 13.4.
8.7	Instructions for Recording and Reporting Occupational Radiation Dose Data	Rev. 2	Nov-05	General	Conforms. Operational program implementation is described in Section 13.4.
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable	Rev. 3	Jun-78	General	Conforms. Operational program implementation is described in Section 13.4.
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	Rev. 1	Jul-93	General	Conforms. Operational program implementation is described in Section 13.4.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable	Rev. 1-R	May-77	General	Conforms. Operational program implementation is described in Section 13.4.
8.11	Applications of Bioassay for Uranium	Rev. 0	Jun-74	General	Not applicable. RG 8.11 has been superseded by RG 8.9, Rev 1.
8.13	Instruction Concerning Prenatal Radiation Exposure	Rev. 3	Jun-99	General	Conforms. Operational program implementation is described in Section 13.4.
8.15	Acceptable Programs for Respiratory Protection	Rev. 1	Oct-99	General	Conforms. Operational program implementation is described in Section 13.4.
8.19	Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants – Design Stage Man-Rem Estimates	Rev. 1	Jun-79	General	Conforms
8.20	Applications of Bioassay for I-125 and I-131	Rev. 1	Sep-79	General	Exception. Per NUREG-1736, RG 8.20 is outdated. RG 8.9 is used. Operational program implementation is described in Section 13.4.
8.25	Air Sampling in the Workplace	Rev. 1	Jun-92	General	Not applicable

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
8.26	Applications of Bioassay for Fission and Activation Products	Rev. 0	Sep-80	General	Exception. Per NUREG-1736, RG 8.26 is outdated. RG 8.9 is used. Operational program implementation is described in Section 13.4.
8.27	Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants	Rev. 0	Mar-81	General	Conforms. Operational program implementation is described in Section 13.4.
8.28	Audible-Alarm Dosimeters	Rev. 0	Jul-81	General	Conforms. Operational program implementation is described in Section 13.4.
8.29	Instruction Concerning Risks from Occupational Radiation Exposure	Rev. 1	Feb-96	General	Conforms. Operational program implementation is described in Section 13.4.
8.32	Criteria for Establishing a Tritium Bioassay Program	Rev. 0	Jul-88	General	Exception. Per NUREG-1736, RG 8.32 is outdated. RG 8.9 is used. Operational program implementation is described in Section 13.4.
8.33	Quality Management Program	Rev. 0	Oct-91	General	Not applicable to nuclear power plants. RG 8.33 applies to nuclear medicine.
8.34	Monitoring Criteria and Methods To Calculate Occupational Radiation Doses	Rev. 0	Jul-92	General	Conforms. Operational program implementation is described in Section 13.4.

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Table 1.9-202 Conformance with Regulatory Guides

RG Number	Title	Revision	Date	RG Position	Evaluation
8.35	Planned Special Exposures	Rev. 0	Jun-92	General	Conforms. Operational program implementation is described in Section 13.4.
8.36	Radiation Dose to the Embryo/Fetus	Rev. 0	Jul-92	General	Conforms. Operational program implementation is described in Section 13.4.
8.38	Control of Access to High and Very High Radiation Areas of Nuclear Plants	Rev. 1	May-06	General	Conforms. Operational program implementation is described in Section 13.4.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.2 1	Introduction and General Description of the Plant	Conforms
C.III.2 1.1	Introduction	Conforms
C.III.2 1.2	General Plant Description	Conforms. Addressed in Sections 1.2.2.19 and 2.0, Figure 2.1-201, and DCD Figures 1.2-1 through 1.2-33.
C.III.2 1.3	Comparisons with Other Facilities	Conforms
C.III.2 1.4	Identification of Agents and Contractors	Conforms
C.III.2 1.5	Requirements for Further Technical Information	Conforms
C.III.2 1.6	Material Referenced	Conforms
C.III.2 1.7	Drawings and Other Detailed Information	Conforms
C.III.2 1.8	Site and Plant Design Interfaces and Conceptual Design Information	Conforms. There are no generic changes or departures from the DCD.
C. III.2 1.9	Conformance with Regulatory Criteria	Conforms
C.III.2 2.1.1	Site Location and Description	Conforms
C.III.2 2.1.2.1	Authority	Conforms
C.III.2 2.1.2.2	Control of Activities Unrelated to Plant Operation	Conforms. There are no known significant changes regarding activities unrelated to plant operation within the exclusion area.
C.III.2 2.1.2.3	Arrangements for Traffic Control	Conforms. There are no known significant changes regarding highways, railroads, or waterways that traverse the exclusion area.
C.III.2 2.1.2.4	Abandonment or Relocation of Roads	Conforms. There are no known significant changes regarding any public roads traversing the exclusion area.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.2 2.1.3	Population Distribution	Conforms
C.III.2 2.2	Nearby Industrial, Transportation, and Military Facilities	Conforms
C.III.2 2.3.1	Regional Climatology	Conforms
C.III.2 2.3.2	Local Meteorology	Conforms
C.III.2 2.3.3	Onsite Meteorological Measurements Program	Conforms. Addressed in SSAR Sections 2.3.3 and 1.8.2 (which commit to RG 1.23, Proposed Revision 1).
C.III.2 2.3.4	Short-Term Atmospheric Dispersion Estimates for Accident Releases	Conforms
C.III.2 2.3.5	Long-Term Atmospheric Dispersion Estimates for Routine Releases	Conforms
C.III.2 2.4.1	Hydrologic Description	Conforms
C.III.2 2.4.2	Floods	Conforms
C.III.2 2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	Conforms
C.III.2 2.4.4	Potential Dam Failures	Conforms
C.III.2 2.4.5	Probable Maximum Surge and Seiche Flooding	Conforms
C.III.2 2.4.6	Probable Maximum Tsunami Hazards	Conforms
C.III.2 2.4.7	Ice Effects	Conforms. Addressed in DCD Appendix 3G.
C.III.2 2.4.8	Cooling Water Canals and Reservoirs	Conforms
C.III.2 2.4.9	Channel Diversions	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.2 2.4.10	Flooding Protection Requirements	Conforms. There are no safety-related SSCs that are not part of the DC facility.
C.III.2 2.4.11	Low Water Considerations	Conforms
C.III.2 2.4.12	Groundwater	Not applicable. A permanent dewatering system is not required.
C.III.2 2.4.13	Accidental Release of Radioactive Liquid Effluent in Ground and Surface Waters	Conforms
C.III.2 2.4.14	Technical Specifications and Emergency Operation Requirements	Conforms
C.III.2 2.5.1	Basic Geologic and Seismic Information	Conforms
C.III.2 2.5.2	Vibratory Ground Motion	Conforms
C.III.2 2.5.3	Surface Faulting	Conforms
C.III.2 2.5.4	Stability of Subsurface Materials and Foundations	Conforms
C.I 2.5.4.1	Geologic Features	Conforms
C.I 2.5.4.2	Properties of Subsurface Materials	Conforms
C.I 2.5.4.3	Foundation Interfaces	Conforms
C.I 2.5.4.4	Geophysical Surveys	Conforms
C.I 2.5.4.5	Excavations and Backfill	Conforms. Addressed in Sections 2.5.4.5 and 17.5.
C.I 2.5.4.6	Ground Water Conditions	Conforms
C.I 2.5.4.7	Response of Soil and Rock to Dynamic Loading	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.I 2.5.4.8	Liquefaction Potential	Conforms
C.I 2.5.4.9	Earthquake Site Characteristics	Conforms
C.I 2.5.4.10	Static Stability	Conforms
C.I 2.5.4.11	Design Criteria	Conforms
C.I 2.5.4.12	Techniques to Improve Subsurface Conditions	Conforms
C.III.2 2.5.5	Stability of slopes	Conforms
C.III.1 3.1	Conformance with NRC General Design Criteria	Conforms. Conformance with the NRC's criteria in 10 CFR 50, Appendix A, is described in DCD Section 3.1 and the applicable DCD system sections.
C.III.1 3.2.1	Seismic Classification	Conforms. There are no additional safety-related or RTNSS SSCs subject to seismic classification beyond those addressed in the DCD. There are no SSCs outside the referenced certified design that are required to be designed for an OBE.
C.III.1 3.2.2	System Quality Group Classification	Conforms. There are no additional safety-related or RTNSS SSCs subject to system quality group classification beyond those addressed in the DCD.
C.III.1 3.3.1 (1)	Wind Loadings	Conforms. There are no safety-related SSCs outside the scope of the certified design. Nonsafety-related facility SSCs that are not included in the referenced certified design meet the requirements of DCD Sections 3.3.1.3 and 3.3.2.3.
C.III.1 3.3.1 (2)	Wind Loadings	Conforms
C.III.1 3.3.2	Tornado Loadings	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.4	Internal Flood Protection	Conforms. There are no SSCs outside the scope of the referenced certified design that require internal flood protection whose failure could prevent a safe shutdown of the plant or result in the uncontrolled release of significant radioactivity.
C.III.1 3.4.2	Analysis Procedures	Conforms. There are no Seismic Category I structures outside the scope of the referenced certified design.
C.III.1 3.5.1.1	Internally Generated Missiles (Outside Containment)	Conforms. There are no SSCs outside the scope of the referenced certified design that are required to be protected against damage from internally generated missiles.
C.III.1 3.5.1.2	Internally Generated Missiles (Inside Containment)	Conforms
C.III.1 3.5.1.3	Turbine Missiles	Conforms. Addressed in DCD Section 10.2.3.8.
C.III.1 3.5.1.4	Missiles Generated by Tornadoes and Extreme Winds	Conforms. Table 2.0-201 demonstrates that the site-specific tornado characteristics are bounded by the parameters assumed in the DCD. DCD Section 3.5.1.4 indicates that resistance to missiles is independent of site topography.
C.III.1 3.5.1.5	Site Proximity Missiles (Except Aircraft)	Conforms
C.III.2 3.5.1.6	Aircraft Hazards	Conforms
C.III.1 3.5.2	Structures, Systems, and Components To Be Protected from Externally Generated Missiles	Conforms. There are no SSCs outside the scope of the referenced certified design that are required to be protected from externally generated missiles.
C.III.1 3.5.3	Barrier Design Procedures	Conforms. There are no SSCs that require reanalysis for tornado, extreme wind, or site proximity missile impact or for aircraft impact.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.6	Protection against Dynamic Effects Associated with the Postulated Rupture of Piping	Conforms
C.III.1 3.6.1	Plant Design for Protection against Postulated Piping Failures in Fluid systems Outside of Containment	Conforms
C.III.1 3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	Conforms
C.III.1 3.6.3	Leak-Before-Break Evaluation Procedures	Not Applicable. ESBWR design does not rely on a Leak Before Break Evaluation.
C.III.1 3.7.1	Seismic Design Parameters	Conforms. Addressed in DCD Sections 3.7 and 3.7.1.
C.III.1 3.7.1.1	Design Ground Motion	Conforms
C.III.1 3.7.1.2	Percentage of Critical Damping Values	Conforms
C.III.1 3.7.1.3	Supporting Media for Seismic Category I Structures	Conforms
C.III.1 3.7.2	Seismic System Analysis	Conforms. Addressed in DCD Section 3.7.2.
C.III.1 3.7.2.1	Seismic Analysis Methods	Conforms
C.III.1 3.7.2.2	Natural Frequencies and Responses	Conforms. Addressed in DCD Section 3.7.2.2.
C.III.1 3.7.2.3	Procedures Used for Analytical Modeling	Conforms
C.III.1 3.7.2.4	Soil/Structure Interaction	Conforms
C.III.1 3.7.2.5	Development of Floor Response Spectra	Conforms. Addressed in DCD Section 3.7.2.5.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.7.2.6	Three Components of Earthquake Motion	Conforms
C.III.1 3.7.2.7	Combination of Modal Responses	Conforms
C.III.1 3.7.2.8	Interaction of Nonseismic Category I Structures with Seismic Category I Structures	Conforms. There are no Seismic Category I structures outside the scope of the referenced certified design. In lieu of providing the plant-specific distances between structures and the heights of structures, the distance and height requirements for Non-Seismic Category I structures are addressed in DCD Section 3.7.2.8.
C.III.1 3.7.2.9	Effects of Parameter Variations on Floor Response Spectra	Conforms. Addressed in DCD Section 3.7.2.9.
C.III.1 3.7.2.10	Use of Constant Vertical Static Factors	Conforms
C.III.1 3.7.2.11	Method Used to Account for Torsional Effects	Conforms
C.III.1 3.7.2.12	Comparison of Responses	Conforms. Addressed in DCD Section 3.7.2.12.
C.III.1 3.7.2.13	Methods for Seismic Analysis of Dams	Not applicable. There are no Seismic Category I dams in the ESBWR design per DCD Section 3.7.3.14.
C.III.1 3.7.2.14	Determination of Dynamic Stability of Seismic Category I Structures	Conforms. Addressed in DCD Sections 3.7.2.14 and 3.8.5.5.
C.III.1 3.7.2.15	Analysis Procedure for Damping	Conforms
C.III.1 3.7.3.1	Seismic Analysis Methods	Conforms
C.III.1 3.7.3.2	Procedures Used for Analytical Modeling	Conforms
C.III.1 3.7.3.3	Analysis Procedure for Damping	Conforms
C.III.1 3.7.3.4	Three Components of Earthquake Motion	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.7.3.5	Combination of Modal Responses	Conforms. Addressed in DCD Section 3.7.3.7.
C.III.1 3.7.3.6	Use of Constant Vertical Static Factors	Conforms
C.III.1 3.7.3.7	Buried Seismic Category I Piping, Conduits, and Tunnels	Conforms. Addressed in DCD Section 3.7.3.13.
C.III.1 3.7.3.8	Methods for Seismic Analysis of Seismic Category I Concrete Dams	Not applicable. There are no Seismic Category I dams for Unit 3.
C.III.1 3.7.3.9	Methods for Seismic Analysis of Above-Ground Tanks	Conforms. Addressed in DCD Section 3.7.3.15.
C.III.1 3.7.4	Seismic Instrumentation	Conforms
C.III.1 3.8.1	Concrete Containment	Conforms
C.III.1 3.8.2	Steel Containment	Conforms
C.III.1 3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	Conforms
C.III.1 3.8.4	Other Seismic Category I Structures	Conforms. There are no Seismic Category I structures that are outside the scope of the DCD.
C.III.1 3.8.5	Foundations	Conforms
C.III.1 3.9.1	Special Topics for Mechanical Components	Conforms. There are no Seismic Category I components or supports beyond those evaluated in the reference certified design.
C.III.1 3.9.1.1	Design Transients	Conforms. There are no Seismic Category I components or supports beyond those evaluated in the reference certified design.
C.III.1 3.9.1.2	Computer Programs Used in Analysis	Conforms. There are no Seismic Category I components or supports beyond those evaluated in the reference certified design.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.9.1.3	Experimental Stress Analysis	Conforms. There are no Seismic Category I components or supports beyond those evaluated in the reference certified design.
C.III.1 3.9.1.4	Considerations for the Evaluation of the Faulted Condition	Conforms. There are no Seismic Category I components or supports beyond those evaluated in the reference certified design.
C.III.1 3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	Conforms. There are no systems outside the scope of the referenced certified design that require dynamic testing and analysis.
C.III.1 3.9.2.1	Piping Vibration, Thermal Expansion, and Dynamic Effects	Conforms. There are no ASME Code Class 1, 2, and 3 systems; other high-energy piping systems inside seismic Category I structures; high-energy portions of systems for which failure could reduce the functioning of any seismic Category I plant feature to an unacceptable level; or seismic Category I portions of moderate-energy piping systems located outside containment outside the scope of the referenced certified design.
C.III.1 3.9.2.2	Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment	Conforms
C.III.1 3.9.2.3	Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions	Conforms. There are no ESBWR pressure vessel internals that the referenced certified design does not cover.
C.III.1 3.9.2.4	Pre-Operational Flow-Induced Vibration Testing of Reactor Internals	Conforms. There are no BWR pressure vessel internals that the referenced certified design does not cover. DCD Sections 3.9.2.3 and 3.9.2.4 adequately cover the analysis of potential adverse flow effects that could impact BWR vessel internals.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.9.2.5	Dynamic System Analysis of the Reactor Internals Under Faulted Condition	Conforms. Addressed in DCD Section 3.9.3.1 and DCD Table 3.9-2.
C.III.1 3.9.2.6	Correlations of Reactor Internals Vibration Tests with the Analytical Results	Conforms. Addressed in DCD Section 3.9.2.6.
C.III.1 3.9.3	ASME Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures	Conforms. There are no pressure-retaining components or component supports designed or constructed in accordance with ASME Code Class 1, 2, or 3, or GDC 1, 2, 4, 14, or 15, beyond those evaluated in the referenced certified design.
C.III.1 3.9.4	Control Rod Drive Systems	Conforms
C.III.1 3.9.5.1	Design Arrangements	Conforms
C.III.1 3.9.5.2	Loading Conditions	Conforms
C.III.1 3.9.5.3	Design Bases	Conforms
C.III.1 3.9.5.4	BWR Reactor Pressure Vessel Internals Including Steam Dryer	Conforms. There are no reactor pressure vessel internals (including the steam dryer) or other main steam system components that are not covered by the referenced certified design. The reactor is classified as non-prototype.
C.III.1 3.9.6.1	Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints	Conforms. There is no safety-related equipment beyond the scope of the referenced certified design.
C.III.1 3.9.6.2	Inservice Testing Program for Pumps	Not applicable. There are no safety-related pumps.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.9.6.3	Inservice Testing Program for Valves	Conforms. Addressed in DCD Section 3.9.6; the list of valves included in the IST program is provided in DCD Table 3.9-8. IST Program test procedures and schedules are addressed in TS Section 5.5.5. Justification for cold shutdown and refueling outage test schedules is addressed in DCD Section 3.9.6 and DCD Table 3.9-8. The implementation milestones for the IST and MOV Programs are addressed in Section 13.4.
C.III.1 3.9.6.3.1	Inservice Testing Program for Motor-Operated Valves (MOVs)	Conforms. Addressed in DCD Section 3.9.6.
C.III.1 3.9.6.3.2	Inservice Testing Program for Power-Operated Valves (POVs) Other Than MOVs	Conforms. Addressed in DCD Section 3.9.6.
C.III.1 3.9.6.3.3	Inservice Testing Program for Check Valves	Conforms. Addressed in DCD Section 3.9.6.
C.III.1 3.9.6.3.4	Pressure Isolation Valve (PIV) Leak Testing	Not applicable. The ESBWR plant does not have any PIVs.
C.III.1 3.9.6.3.5	Containment Isolation Valve (CIV) Leak Testing	Conforms
C.III.1 3.9.6.3.6	Inservice Testing Program for Safety and Relief Valves	Conforms. Addressed in DCD Table 3.9-8.
C.III.1 3.9.6.3.7	Inservice Testing Program for Manually Operated Valves	Conforms. Addressed in DCD Table 3.9-8.
C.III.1 3.9.6.3.8	Inservice Testing Program for Explosively Activated Valves	Conforms. Addressed in DCD Table 3.9-8.
C.III.1 3.9.6.4	Inservice Testing Program for Dynamic Restraints	Conforms with the following exception: A plant specific snubber table will be prepared in conjunction with closure of ITAAC Table 3.1-1.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.9.6.5	Relief Requests and Alternative Authorizations to ASME OM Code	Conforms
C.III.1 3.10.1	Seismic Qualification Criteria	Conforms. There is no seismic or dynamic qualification required for equipment that is outside the scope of the referenced certified design.
C.III.1 3.10.2	Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation	Conforms
C.III.1 3.10.3	Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation	Conforms
C.III.1 3.10.4	Test and Analyses Results and Experience Database	Conforms
C.III.1 3.11	Environmental Qualification of Mechanical and Electrical Equipment	Conforms. There is no other equipment beyond that which has been evaluated in the referenced certified design.
C.III.1 3.11.1	Equipment Location and Environmental Conditions	Conforms
C.III.1 3.11.2	Qualification Tests and Analysis	Conforms
C.III.1 3.11.3	Qualification Test Results	Conforms
C.III.1 3.11.4	Loss of Ventilation	Conforms
C.III.1 3.11.5	Estimated Chemical and Radiation Environment	Conforms
C.III.1 3.11.6	Qualification of Mechanical Equipment	Conforms
C.III.1 3.12.1	Introduction	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.12.2	Codes and Standards	Conforms. Addressed in DCD Sections 3.2, 3.6, 3.7, and in Chapters 5 and 14.
C.III.1 3.12.3	Piping Analysis Methods	Conforms. Addressed in DCD Sections 3.7.2.2 and 3.7.3.9.
C.III.1 3.12.3.1	Experimental Stress Analyses	Conforms. Addressed in DCD Section 3.9.1.3.
C.III.1 3.12.3.2	Modal Response Spectrum Method	Conforms. Addressed in DCD Section 3.7.2.1.
C.III.1 3.12.3.3	Response Spectra Method (or Independent Support Motion Method)	Conforms. Addressed in DCD Section 3.7.2.1.2.
C.III.1 3.12.3.4	Time History Method	Conforms. Addressed in DCD Section 3.7.2.1.1.
C.III.1 3.12.3.5	Inelastic Analyses Method	Not Applicable. Per DCD Section 3.9.1.4 (Inelastic Analyses Methods), except for pipe whip restraints, inelastic analyses methods are not used in the ESBWR piping design and analysis.
C.III.1 3.12.3.6	Small-Bore Piping Method	Conforms. Addressed in DCD Section 3.7.3.16.
C.III.1 3.12.3.7	Nonseismic/Seismic Interaction (II/I)	Conforms with the following exception: The location and distance between piping systems will be established as part of the completion of ITAAC Table 3.1-1.
C.III.1 3.12.3.8	Seismic Category I Buried Piping	Not Applicable. Per DCD Section 3.7.3.13, there is no buried Seismic Category I piping.
C.III.1 3.12.4	Piping Modeling Technique	Conforms. Addressed in DCD Section 3.7.3.3.1 and Appendix 3D for the PISYS computer code.
C.III.1 3.12.4.1	Computer Codes	Conforms. Addressed in DCD Appendix 3D.
C.III.1 3.12.4.2	Dynamic Piping Model	Conforms. Addressed in DCD Section 3.7.3.3.1.
C.III.1 3.12.4.3	Piping Benchmark Program	Conforms. Addressed in DCD Appendix 3D.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.12.4.4	Decoupling Criteria	Conforms. Addressed in DCD Sections 3.7.2.3 and 3.7.3.16.
C.III.1 3.12.5.1	Seismic Input Envelope vs. Site-Specific Spectra	Conforms. Addressed in DCD Section 3.7.1.
C.III.1 3.12.5.2	Design Transients	Conforms. Addressed in DCD Section 3.9.1.1 and DCD Table 3.9-1.
C.III.1 3.12.5.3	Loadings and Load Combination	Conforms. Addressed in DCD Section 3.9.1.1 and DCD Table 3.9-8.
C.III.1 3.12.5.4	Damping Values	Conforms. Addressed in DCD Section 3.7.1.2 and DCD Table 3.7-1.
C.III.1 3.12.5.5	Combination of Modal Responses	Conforms. Addressed in DCD Section 3.7.3.7.
C.III.1 3.12.5.6	High-Frequency Modes	Conforms. Addressed in DCD Sections 3.7.1.1 and 3.7.1.2.
C.III.1 3.12.5.7	Fatigue Evaluation of ASME Code Class 1 Piping	Conforms. Addressed in DCD Section 3.9.3.4 and DCD Table 3.9-8.
C.III.1 3.12.5.8	Fatigue Evaluation of ASME Code Class 2 and 3 Piping	Conforms. Addressed in DCD Section 3.9.
C.III.1 3.12.5.9	Thermal Oscillations in Piping Connected to the Reactor Coolant System	Conforms
C.III.1 3.12.5.10	Thermal Stratification	Conforms. Addressed in DCD Section 3.9.2.1.2.
C.III.1 3.12.5.11	Safety Relief Valve Design, Installation, and Testing	Conforms. Addressed in DCD Figures 5.2-3 and 5.4-3, and DCD Table 3.9-8.
C.III.1 3.12.5.12	Functional Capability	Conforms. Addressed in DCD Table 3.9-2, Note 13, and DCD Chapters 5 and 6.
C.III.1 3.12.5.13	Combination of Inertial and Seismic Anchor Motion Effects	Conforms. Addressed in DCD Section 3.7.3.9.
C.III.1 3.12.5.14	Operating-Basis Earthquake as a Design Load	Not applicable. The SSE establishes the design load for the ESBWR.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.12.5.15	Welded Attachments	Conforms. Addressed in DCD Section 3.9.3.7.1.
C.III.1 3.12.5.16	Modal Damping for Composite Structures	Conforms. Addressed in DCD Section 3.7.2.13.
C.III.1 3.12.5.17	Minimum Temperature for Thermal Analyses	Conforms. Addressed in DCD Sections 3.9.1.1 and 3.9.3.1.
C.III.1 3.12.5.18	Intersystem Loss-of-Coolant Accident	Conforms. Addressed in DCD Appendix 3K.
C.III.1 3.12.5.19	Effects of Environment on Fatigue Design	Conforms. Addressed in DCD Section 3.9.3.4. The reference in RG 1.206 to 1.76 appears to be in error, and should have referenced 1.207.
C.III.1 3.12.6.1	Applicable Codes	Conforms. Addressed in DCD Section 3.9.3.7.1.
C.III.1 3.12.6.2	Jurisdictional Boundaries	Conforms. Addressed in DCD Section 3.9.3.7.1.
C.III.1 3.12.6.3	Loads and Load Combinations	Conforms. Addressed in DCD Section 3.9 and DCD Appendix 3B.
C.III.1 3.12.6.4	Pipe Support Baseplate and Anchor Bolt Design	Conforms. Addressed in DCD Section 3.9.3.7.
C.III.1 3.12.6.5	Use of Energy Absorbers and Limit Stops	Conforms. Addressed in DCD Section 3.9.3.7.
C.III.1 3.12.6.6	Use of Snubbers	Conforms. Addressed in DCD Section 3.9.3.7.1(3).
C.III.1 3.12.6.7	Pipe Support Stiffnesses	Conforms. Addressed in DCD Section 3.9.3.7.1.
C.III.1 3.12.6.8	Seismic Self-Weight Excitation	Conforms. Addressed in DCD Section 3.9.3.7.1.
C.III.1 3.12.6.9	Design of Supplementary Steel	Conforms. Addressed in DCD Section 3.9.3.7.1.
C.III.1 3.12.6.10	Consideration of Friction Forces	Conforms. Addressed in DCD Section 3.9.3.7.1(5).
C.III.1 3.12.6.11	Pipe Support Gaps and Clearances	Conforms. Addressed in DCD Section 3.9.3.7.1.
C.III.1 3.12.6.12	Instrumentation Line Support Criteria	Conforms. Addressed in DCD Section 3.9.3.7.1.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 3.12.6.13	Pipe Deflection Limits	Conforms. Addressed in DCD Section 3.9.2.1.1 and Chapter 14.
C.III.1 3.13	Threaded Fasteners – ASME code Class 1, 2, and 3	Conforms
C.III.1 3.13.1.1	Materials Selection	Conforms
C.III.1 3.13.1.2	Special Materials Fabrication Processes and Special Controls	Conforms
C.III.1 3.13.1.3	Fracture Toughness Requirements for Threaded Fasteners Made of Ferritic Materials	Conforms
C.III.1 3.13.1.5	Certified Material Test Reports	Conforms
C.III.1 3.13.2	Inservice Inspection Requirements	Conforms
C.III.1 4.1	Reactor: Summary Description	Conforms
C.III.1 4.2	Fuel System Design	Conforms
C.III.1 4.3	Nuclear Design	Conforms
C.III.1 4.4	Thermal and Hydraulic Design	Conforms
C.III.1 4.5.1	Control Rod Drive Structural Materials	Conforms
C.III.1 4.5.2	Reactor Internal and Core Support Materials	Conforms
C.III.1 4.6	Functional Design of Reactivity Control System	Conforms
C.III.1 5.1	Reactor Coolant and Connecting Systems: Summary Description	Conforms
C.III.1 5.2.1	Compliance with ASME Codes and Code Cases	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 5.2.2.1	Design Bases	Conforms
C.III.1 5.2.2.2	Design Evaluation	Conforms
C.III.1 5.2.2.3	Piping and Instrumentation Diagrams	Conforms
C.III.1 5.2.2.4	Equipment and Component Description	Conforms
C.III.1 5.2.2.5	Mounting of Pressure-Relief Devices	Conforms
C.III.1 5.2.2.6	Applicable Codes and Classification	Conforms
C.III.1 5.2.2.7	Material Specification	Conforms
C.III.1 5.2.2.8	Process Instrumentation	Conforms
C.III.1 5.2.2.9	System Reliability	Conforms
C.III.1 5.2.2.10	Testing and Inspection	Conforms. Addressed in DCD Section 5.2.2.4, and in Section 3.9 and Chapter 14.
C.III.1 5.2.3.1	Material Specifications	Conforms
C.III.1 5.2.3.2	Compatibility with Reactor Coolant	Conforms. Addressed in DCD Section 5.2.3.
C.III.1 5.2.3.3	Fabrication and Processing of Ferritic Materials	Conforms
C.III.1 5.2.3.4	Fabrication and Processing of Austenitic Stainless Steels	Conforms
C.III.1 5.2.3.5	Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Based Alloys (PWRs only)	Not applicable. Applies only to PWRs.
C.III.1 5.2.3.6	Threaded Fasteners	Conforms. Addressed in DCD Section 3.9.3.9.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 5.2.4.1	Inservice Inspection and Testing Program	Conforms. Addressed in DCD Section 5.2.4 and in Section 5.2.4.
C.III.1 5.2.4.2	Preservice Inspection and Testing Program	Conforms. Addressed in DCD Section 5.2.4.
C.III.1 5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	Conforms
C.III.1 5.3.1.1	Material Specifications	Conforms
C.III.1 5.3.1.2	Special Processes Used for Manufacturing and Fabrication	Conforms
C.III.1 5.3.1.3	Special Methods for Nondestructive Examination	Conforms
C.III.1 5.3.1.4	Special Controls for Ferritic and Austenitic Stainless Steels	Conforms
C.III.1 5.3.1.5	Fracture Toughness	Conforms
C.III.1 5.3.1.6	Material Surveillance	Conforms. Addressed in DCD Section 5.3.1.6 and Section 5.3.1.8.
C.III.1 5.3.1.7	Reactor Vessel Fasteners	RG does not contain any guidance in this section.
C.III.1 5.3.2.1	Limit Curves	Conforms
C.III.1 5.3.2.2	Operating Procedures	Conforms. Addressed in DCD Sections 5.3.2.1, 5.3.2.2, and 5.3.3.6, and in Section 5.3.3.6.
C.III.1 5.3.2.3	Pressurized Thermal Shock (PWRs only)	Not applicable. Applies only to PWRs.
C.III.1 5.3.2.4	Upper-Shelf Energy	Conforms
C.III.1 5.3.3	Reactor Vessel Integrity	Conforms. Identification of a specific manufacturer is not required.
C.III.1 5.3.3.1	Design	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 5.3.3.2	Materials of Construction	Conforms
C.III.1 5.3.3.3	Fabrication Methods	Conforms
C.III.1 5.3.3.4	Inspection Requirements	Conforms. Addressed in DCD Section 5.3.3.4.
C.III.1 5.3.3.5	Shipment and Installation	Conforms. Addressed in DCD Section 5.3.3.5.
C.III.1 5.3.3.6	Operating Conditions	Conforms. Addressed in DCD Section 5.3.3.6.
C.III.1 5.3.3.7	Inservice Surveillance	Conforms. Addressed in DCD Section 5.3.3.7.
C.III.1 5.3.3.8	Threaded Fasteners	Conforms. Addressed in DCD Section 3.9.3.9.
C.III.1 5.4.1	Reactor Coolant Pumps or Circulation Pumps (BWR)	Conforms
C.III.1 5.4.1.1	Pump Flywheel Integrity (PWR)	Not applicable. Applies only to PWRs.
C.III.1 5.4.2	Steam Generators (PWR)	Not applicable. Applies only to PWRs.
C.III.1 5.4.3	Reactor Coolant System Piping and Valves	Conforms
C.III.1 5.4.4	Main Steamline Flow Restrictions	Conforms
C.III.1 5.4.5	Pressurizer	Not applicable. Applies only to PWRs.
C.III.1 5.4.6	Reactor Core Isolation Cooling System (BWRs/Isolation Condenser System (Economic Simplified BWR)	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 5.4.7	Residual Heat Removal System/Passive Residual Heat Removal System (Advanced Light-Water Reactor) Shutdown Cooling Mode of the Reactor Water Cleanup System (Economic Simplified BWR)	Conforms
C.III.1 5.4.8	Reactor Water Cleanup System (BWR) Reactor Water Cleanup/Shutdown Cooling System (Economic Simplified BWR)	Conforms
C.III.1 5.4.9	Reactor Coolant System Pressure Relief Devices/Reactor Coolant Depressurization Systems	Conforms
C.III.1 5.4.10	Reactor Coolant System Component Supports	Conforms
C.III.1 5.4.11	Pressurizer Relief Discharge System (PWRs only)	Not applicable. Applies only to PWRs.
C.III.1 5.4.12	Reactor Coolant System High-Point Vents	Conforms
C.III.1 5.4.13	Main Steamline, Feedwater, and Auxiliary Feedwater Piping	Conforms
C.III.1 6.1	Engineered Safety Features: Engineered Safety Feature Materials	Conforms. Addressed in DCD Section 6.1.
C.III.1 6.1.1.1	Materials Selection and Fabrication	Conforms
C.III.1 6.1.1.2	Composition and Compatibility of Core Cooling Coolants and Containment Sprays	Conforms. Addressed in DCD Sections 5.2.3.2, 5.2.3.4.1, 5.4.8, 6.1.3.4, 6.1.1.4, 6.1.2, 9.1.3, and 9.3.10.
C.III.1 6.1.2	Organic Materials	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 6.2	Containment Systems	Conforms
C.III.1 6.2.1	Containment Functional Design	Conforms
C.III.1 6.2.2	Containment Heat Removal Systems	Conforms
C.III.1 6.2.3	Secondary Containment Functional Design	Not Applicable. The ESBWR plant does not have a secondary containment.
C.III.1 6.2.4	Containment Isolation System	Conforms.
C.III.1 6.2.5	Combustible Gas Control in Containment	Conforms.
C.III.1 6.2.6	Containment Leakage Testing	Conforms. Addressed in DCD Sections 6.2.6.1 through 6.2.6.4, and in Section 13.4. Special testing requirements in RG 1.206, Section C.III.1, Section 6.2.6.5 are not applicable to the ESBWR.
C.III.1 6.2.7	Fracture Prevention of Containment Pressure Vessel	Conforms
C.III 6.3	Emergency Core Cooling System	Conforms. There are no aspects of the site-specific design that affect the LOCA analyses in the DCD.
C.III.1 6.4	Habitability Systems	Conforms
C.III.2 6.5	Fission Product Removal and Control Systems	Conforms
C.III.1 6.6	Inservice Inspection of Class 2 and 3 Components	Conforms. Addressed in DCD Section 6.6 and in Section 6.6.10.3.
C.III.1 6.6.1	Components Subject to Examination	Conforms
C.III.1 6.6.2	Accessibility	Conforms
C.III.1 6.6.3	Examination Techniques and Procedures	Conforms. Addressed in DCD Section 6.6.3.2. There are no special examination techniques required to meet the ASME Code.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 6.6.4	Inspection Intervals	Conforms. Addressed in DCD Section 6.6.4.
C.III.1 6.6.5	Examination Categories and Requirements	Conforms. Addressed in DCD Section 6.6.3.1.
C.III.1 6.6.6	Evaluation of Examination Results	Conforms (addressed in DCD Section 6.6.5), except that RG 1.206 references ASME Code Sections IWC-4000 and IWD-4000 for Class 2 and Class 3, respectively, whereas DCD Section 6.6.5 references IWA-4000. Later editions of ASME Code Section XI do not contain Sections IWC-4000 and IWD-4000, only IWA-4000. Therefore, the intent of the RG is met.
C.III.1 6.6.7	System Pressure Tests	Conforms. Addressed in DCD Section 6.6.6.
C.III.1 6.6.8	Augmented Inservice Inspection to Protect against Postulated Piping Failures	Conforms. Addressed in DCD Section 6.6.7.
C.III.1 6.7	Main Steamline Isolation Valve Leakage Control Steam (BWRs)	Not applicable to the ESBWR.
C.III.1 7	Instrumentation and Controls	Conforms. Addressed in DCD Chapter 7, Tier 1, and design-related ITAAC (DAC). There are no departures from the referenced certified design.
C.III.1 7.1	Introduction	Conforms. There is no safety-related instrumentation, control, or supporting system that has not been addressed in the referenced certified design or other parts of the COL application.
C.III.1 7.2	Reactor Trip System	Conforms. There is no reactor trip system instrumentation, control, or supporting system that has not been addressed in the referenced certified design or other parts of the COL application.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 7.3	Engineered Safety Features Systems	Conforms. There are no ESF systems I&C or supporting systems that have not been addressed in the referenced certified design or other parts of the COL application.
C.III.1 7.4	Systems Required for safe Shutdown	Conforms. There are no safe-shutdown systems I&C or supporting systems that have not been addressed in the referenced certified design or other parts of the COL application.
C.III.1 7.5	Information Systems Important to Safety	Conforms. There are no information systems important to safety that have not been addressed in the referenced certified design or other parts of the COL application.
C.III.1 7.6	Interlock Systems Important to Safety	Conforms. There are no interlock systems important to safety that have not been addressed in the referenced certified design or other parts of the COL application.
C.III.1 7.7	Control Systems Not Required for Safety	Conforms. There is no control system instrumentation or supporting system that has not been addressed in the referenced certified design or other parts of the COL application.
C.III.1 7.8	Diverse Instrumentation and Control Systems	Conforms. There is no diverse I&C system that has not been addressed in the referenced certified design or other parts of the COL application.
C.III.1 7.9	Data Communication Systems	Conforms. There are no data communication systems that have not been addressed in the referenced certified design or other parts of the COL application.
C.III.1 8	Electrical Power	Conforms
C.III.1 8.1	Introduction	Conforms. There are no safety-related or RTNSS onsite AC or DC loads that are added to the referenced certified design. There are no safety-related or RTNSS electrical systems that are beyond the scope of the referenced certified design.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 8.2.1	Description	Conforms. Addressed in Section 8.2.
C.III.1 8.2.2	Analysis	Conforms. Addressed in Section 8.2.
C.III.1 8.3.1.1	AC Power Systems: Description	Conforms. Addressed in DCD Section 8.3.1 and in Section 8.3.1.1.
C.III.1 8.3.1.2	Analysis	Not applicable. Does not request information for passive designs.
C.III.1 8.3.1.3	Electrical Power System Calculations and Distribution System Studies for AC Systems	Conforms
C.III.1 8.3.2.1	DC Power Systems: Description	Not applicable. Does not request information for passive designs.
C.III.1 8.3.2.2	Analysis	Not applicable. Does not request information for passive designs.
C.III.1 8.3.2.3	Electrical Power System Calculations and Distribution System Studies for DC Systems	Conforms
C.III.1 8.4.1(1)	Station Blackout: Description	Not applicable. Does not request information for passive designs.
C.III.1 8.4.1(2)		Not applicable. Does not request information for passive designs.
C.III.1 8.4.1(3)		Conforms. Addressed in Section 8.3.2.1.1.
C.III.1 8.4.1(4)		Conforms. Addressed in Section 8.3.2.1.1.
C.III.1 8.4.2	Analysis	Not applicable. Does not request information for passive designs.
C.III 9.1.1	Fuel Storage and Handling: Criticality Safety of Fresh and Spent Fuel Storage and Handling	Conforms. Addressed in DCD Sections 9.1.1 and 9.1.2.
C.III 9.1.2	New and Spent Fuel Storage	Conforms. Addressed in DCD Section 9.1.2.
C.III 9.1.3	Spent Fuel Pool Cooling and Cleanup System	Conforms. Addressed in DCD Section 9.1.3.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III 9.1.4	Light Load Handling System (Related to Refueling)	Conforms
C.III.1 9.1.5	Overhead Heavy Load Handling System	Conforms. Addressed in DCD Section 9.1.5.5 and in Sections 9.1.4 and 9.1.5.
C.III.1 9.2.1.1	Station Service Water System (Open, Raw Water Cooling Systems): Design Bases	Conforms. Addressed in DCD Section 9.2.1.1.
C.III.1 9.2.1.2	System Description	Conforms. Addressed in DCD Section 9.2.1.2 and in Section 9.2.1.2.
C.III.1 9.2.1.3	Safety Evaluation	Conforms. Addressed in DCD Section 9.2.1.3 and in Section 9.2.1.2 (for long-term corrosion and fouling).
C.III.1 9.2.1.4	Inspection and Testing Requirements	Conforms. Addressed in DCD Section 9.2.1.4.
C.III.1 9.2.1.5	Instrumentation Requirements	Conforms. Addressed in DCD Section 9.2.1.5.
C.III 9.2.2	Cooling System for Reactor Auxiliaries (Closed Cooling Water Systems)	Conforms
C.III.1 9.2 (for DCD Section 9.2.3)	Makeup Water System Design Bases	Conforms. Design Bases, Safety Evaluation, Inspection and Testing Requirements, and Instrumentation are addressed in DCD Section 9.2.3. System Description is addressed in Section 9.2.3.
C.III.1 9.2.4	Potable and Sanitary Water Systems Design Bases	Conforms
C.III.1 9.2.5	Ultimate Heat Sink	The design of the UHS is within the scope of the referenced certified design, and inspection and testing requirements are addressed in DCD Section 9.2.5.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 9.2.6	Condensate Storage Facilities	Conforms. There are no safety-related or RTNSS condensate storage facilities outside the scope of the referenced certified design that are sources of water for residual heat removal or sources of coolant inventory makeup for safety-related systems.
C.III.1 9.2 (for DCD Section 9.2.7)	Chilled Water System	Conforms. Addressed in DCD Section 9.2.7.
C.III.1 9.2 (for DCD Section 9.2.8)	Turbine Component Cooling Water System	Conforms. Addressed in DCD Section 9.2.8.
C.III.1 9.2 (for DCD Section 9.2.10)	Station Water System	Conforms. Design Bases, Safety Evaluation, Inspection and Testing Requirements, and Instrumentation are addressed in DCD Section 9.2.10. System Description is addressed in Section 9.2.10.
C.III.1 9.3	Process Auxiliaries	Conforms. Hydrogen Water Chemistry is addressed in Section 9.3.9, Oxygen Injection System is addressed in Section 9.3.10, Zinc Injection System is addressed in Section 9.3.11, and Auxiliary Boiler System is addressed in DCD Section 9.3.12.
C.III.1 9.3 1	Compressed Air Systems	Conforms. Instrument Air is addressed in DCD Section 9.3.6, Service Air is addressed in DCD Section 9.3.7, and High Pressure Nitrogen Supply System is addressed in DCD Section 9.3.8.
C.III.1 9.3.2	Process and Postaccident Sampling Systems	Conforms
C.III.1 9.3.3	Equipment and Floor Drain System	Conforms. Addressed in DCD Section 9.3.3.
C.III.1 9.3.4	Chemical and Volume Control System (PWRs) (Including Boron Recovery System)	Not applicable. Applies only to PWRs.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 9.3.5	Standby Liquid Control System	Conforms
C.III.1 9.4	Air Conditioning, Heating, Cooling, and Ventilation Systems	Conforms. Reactor Building HVAC System is addressed in DCD Section 9.4.6, Electric Building Heating, Ventilation, and Air Conditioning System is addressed in DCD Section 9.4.7, and Drywell Cooling System is addressed in DCD Section 9.4.8.
C.III.1 9.4.1	Control Room Area Ventilation System	Conforms
C.III.1 9.4.2	Spent Fuel Pool Area Ventilation Systems	Conforms
C.III.1 9.4.3	Auxiliary and Radwaste Area Ventilation System	Conforms
C.III.1 9.4.4	Turbine Building Area Ventilation System	Conforms
C.III.1 9.4.5	Engineered Safety Feature Ventilation System	Conforms
C.III.I 9.5.1	Fire Protection Program	Conforms
C.III.1 9.5.1.1(1)		Conforms
C.III.1 9.5.1.1(2)		Conforms
C.III.1 9.5.1.1(3)		Conforms. Addressed in DCD Section 1.7.
C.III.1 9.5.1.1(4)		Conforms. Will be completed in accordance with the milestones in Section 13.4.
C.III.1 9.5.1.1(5)		Conforms. Will be completed in accordance with the milestones in Section 13.4.
C.III.1 9.5.1.1(6)		Conforms
C.III.1 9.5.1.1(7)		Conforms. Will be completed in accordance with the milestones in Section 13.4.

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Section	Section Title	Conformance Evaluation
C.III.1 9.5.1.1(8)		Conforms
C.III.1 9.5.1.1(9)		Conforms. Addressed in DCD Sections 9.5.1.15 and 14.3, and in Section 13.4.
C.III.1 9.5.2	Communication System	Conforms. Addressed in DCD Section 9.5.2 and in Section 9.5.2.
C.III.1 9.5.3	Lighting System	Conforms. Addressed in DCD Section 9.5.3.
C.III.1 9.5.4	Diesel Generator Fuel Oil Storage and Transfer Systems	Conforms. Addressed in DCD Section 9.5.4 and in Section 9.5.4.
C.III.1 9.5.4.1	Design Basis	Conforms. Addressed in DCD Section 9.5.4.
C.III.1 9.5.4.2	System Description	Conforms
C.III.1 9.5.4.3	Safety Evaluation	Conforms
C.III.1 9.5.5	Diesel Generator Cooling Water Systems	Conforms. Addressed in DCD Section 9.5.5.
C.III.1 9.5.6	Diesel Generator Starting Systems	Conforms. Addressed in DCD Section 9.5.6.
C.III.1 9.5.7	Diesel Generator Lubrication Systems	Conforms. Addressed in DCD Section 9.5.7.
C.III.1 9.5.8	Diesel Generator Combustion Air Intake and Exhaust System	Conforms. Addressed in DCD Section 9.5.8.
C.III.1 10.1	Steam and Power Conversion: Introduction	Conforms. There are no principal design features of the steam and power conversion system that are outside the scope of the referenced certified design.
C.III.1 10.2.1 (1)	Design Bases	Conforms. Addressed in DCD Section 10.2.1.
C.III.1 10.2.1 (2)	Design Bases	Conforms. Addressed in DCD Section 10.2.2.
C.III.1 10.2.1 (3)	Design Bases	Conforms. Addressed in DCD Sections 3.5.1, 3.5.3, 3.6, and 10.2.4, and DCD Figure 3.5-2.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 10.2.2 (1)	Description	Conforms. Addressed in DCD Sections 10.2.2, 10.2.3, and DCD Figures 1.2-12 to 1.2-20, 3.5-2, and 10.1-1.
C.III.1 10.2.2 (2)	Description	Conforms. Addressed in DCD Sections 10.2.2 and 10.2.3.
C.III.1 10.2.2 (3)	Description	Conforms. Addressed in DCD Section 10.2.2 and DCD Figures 10.2-1, 10.2-2, and 10.2-3.
C.III.1 10.2.2 (4)	Description	Conforms. Addressed in DCD Sections 10.2.3 and 14.2.8.
C.III.1 10.2.2 (5)	Description	Conforms. Addressed in DCD Sections 12.2.1, 12.2.3, 12.4.4, DCD Table 12.2-23, and DCD Figures 12.3-12 to 12.3-18 and 12.3-32 to 12.3-38.
C.III.1 10.2.2 (6)	Description	Conforms. Addressed in DCD Sections 3.6, 10.2.2, and 10.2.4.
C.III.1 10.2.3 (1)	Turbine Rotor Integrity	Conforms. Addressed in DCD Section 10.2.3 and Section 10.2.3.8.
C.III.1 10.2.3 (2)	Turbine Rotor Integrity	Conforms. Addressed in DCD Section 10.2.3 and Section 10.2.3.8.
C.III.1 10.2.3 (3)	Turbine Rotor Integrity	Conforms. Addressed in DCD Section 10.2.3 and Section 10.2.3.8.
C.III.1 10.2.3 (4)	Turbine Rotor Integrity	Conforms. Addressed in DCD Section 10.2.3 and Section 10.2.3.8.
C.III.1 10.2.3 (5)	Turbine Rotor Integrity	Conforms. Addressed in DCD Sections 10.2.2 and 10.2.3, and Section 10.2.3.8.
C.III.1 10.3	Main Steam Supply System	Conforms. Addressed in DCD Section 10.3.
C.III.1 10.3.1 (1)	Design Bases	Conforms. Addressed in DCD Section 10.3.1.
C.III.1 10.3.1 (2)	Design Bases	Conforms. Addressed in DCD Section 10.3.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 10.3.1 (3)	Design Bases	Conforms. Addressed in DCD Sections 10.3.2 and 10.3.3.
C.III.1 10.3.1 (4)	Design Bases	Conforms. Addressed in DCD Section 10.3.
C.III.1 10.3.1 (5)	Design Bases	Conforms. Addressed in DCD Section 10.3.
C.III.1 10.3.1 (6)	Design Bases	Conforms. Addressed in DCD Section 10.3.
C.III.1 10.3.2	Description	Conforms. Addressed in DCD Section 10.3.
C.III.1 10.3.3	Evaluation	Conforms. Addressed in DCD Section 10.3.
C.III.1 10.3.4	Inspection and Testing Requirements	Conforms. Addressed in DCD Section 10.3.4.
C.III.1 10.3.5	Water Chemistry (PWR Only)	Not applicable. Only applies to PWRs.
C.III.1 10.3.6 (1)	Steam and Feedwater System Materials	Conforms. Addressed in DCD Section 10.3.6.
C.III.1 10.3.6 (2)	Steam and Feedwater System Materials	Conforms. Addressed in DCD Sections 6.6 and 10.3.4.
C.III.1 10.3.6 (3)	Steam and Feedwater System Materials	Not applicable. DCD Section 10.3.6 states that there are no austenitic stainless steels in the steam and feedwater system piping.
C.III.1 10.3.6 (4)	Steam and Feedwater System Materials	Not Applicable. DCD Section 10.3.6 states that there are no austenitic stainless steels in the ASME Code Section III Class 1 and 2 portions of steam and feedwater piping.
C.III.1 10.3.6 (5)	Steam and Feedwater System Materials	Conforms. Addressed in DCD Section 10.3.
C.III.1 10.3.6 (6)	Steam and Feedwater System Materials	Not applicable
C.III.1 10.4 (1)	Other Features of the Steam and Power Conversion System	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 10.4.1	Main Condensers	Conforms. Sampling points for detection are discussed in DCD Section 10.4.1.5.4. Although sodium content and sampling for sodium content is not specifically mentioned in DCD Section 10.4.1, monitoring condensate for an increase in conductivity is considered an acceptable means to detect condenser tube leakage. A table of key parameters and associated action levels is provided as Table 10.4-201. Alarm setpoints are established to provide an indication of abnormal chemistry conditions prior to reaching a recommended action level.
C.III.1 10.4.2	Main Condenser Evacuation System	Conforms. There are no design features of the main condenser evacuation system that are outside the scope of the referenced certified design.
C.III.1 10.4.3 (1)	Turbine Gland Sealing System	Conforms. Addressed in DCD Section 10.4.3.
C.III.1 10.4.3 (2)		Conforms with the following exception: For the operational phase, the QA Program is described in Chapter 17, and is based on NQA-1, rather than RG 1.33.
C.III.1 10.4.4 (1)	Turbine Bypass System	Conforms. The Turbine Bypass System is consistent with the referenced certified design.
C.III.1 10.4.5 (1)	Circulating Water System	Conforms
C.III.1 10.4.5 (2)		Not applicable. The circulating water system does not interface with the UHS.
C.III.1 10.4.6 (1)	Condensate Cleanup System	Conforms
C.III.1 10.4.6 (2)		Conforms. Addressed in DCD Sections 10.4.1, 10.4.6, and 5.2.3, DCD Table 5.2-5, and in Table 10.4-201.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 10.4.6 (3)		Conforms
C.III.1 10.4.6 (4)		Not applicable. Only applies to PWRs.
C.III.1 10.4.7 (1)	Condensate and Feedwater Systems	Not applicable. Only applies to PWRs.
C.III.1 10.4.7 (2)		Conforms. Addressed in DCD Sections 1.2.2 and 5.2.4, and DCD Tables 1.9-22 and 1.11-1.
C.III.1 10.4.7 (3)		Not applicable. The condensate and feedwater systems are consistent with the referenced certified design.
C.III.1 10.4.8	Steam Generator Blowdown System (PWR)	Not applicable. Only applies to PWRs.
C.III.1 10.4.9	Auxiliary Feedwater System (PWR)	Not applicable. Only applies to PWRs.
C,III.1 11.1	Source Terms	Conforms
C.III.1 11.2.1(1)	Liquid Waste Management Systems: Design Bases	Conforms. Addressed in DCD Section 11.2 and in Section 11.2.
C.III.1 11.2.1(2)	Design Bases	Conforms. Addressed in DCD Section 11.2.
C.III.1 11.2.1(3)	Design Bases	Conforms. Addressed in DCD Section 11.2.1 and DCD Table 11.2-3. Conformance with RG 1.140 is addressed in DCD Section 9.4.3.
C.III.1 11.2.1(4)	Design Bases	Conforms. Addressed in DCD Section 9.4.3.
C.III.1 11.2.1(5)	Design Bases	Conforms. Addressed in DCD Sections 11.2.3 and 15.3.16 and in Section 2.4.13.
C.III.1 11.2.1(6)	Design Bases	Conforms. Quality Assurance Program requirements are addressed in Chapter 17.
C.III.1 11.2.1(7)	Design Bases	Conforms. Addressed in DCD Section 11.2.4.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 11.2.1(8)	Design Bases	Conforms
C.III.1 11.2.1(9)	Design Bases	Conforms. Addressed in DCD Section 11.2.2 and in Section 11.2.
C.III.1 11.2.2(1)	System Description	Conforms. Addressed in DCD Section 11.2.2.
C.III.1 11.2.2(2)	System Description	Conforms. Addressed in DCD Section 11.2.2.
C.III.1 11.2.2(3)	System Description	Conforms. Addressed in DCD Section 11.2.2.
C.III.1 11.2.2(4)	System Description	Conforms. Addressed in DCD Section 11.2.2.
C.III.1 11.2.3(1)	Radioactive Effluent Releases	Conforms. Addressed in DCD Sections 11.2 and 12.2, and in Section 12.2.
C.III.1 11.2.3(2)	Radioactive Effluent Releases	Conforms. Addressed in DCD Sections 11.2 and 12.2, and in Section 12.2.
C.III.1 11.3.1(1)	Gaseous Waste Management Systems: Design Bases	Addressed in DCD Section 11.3. Conforms with the following exception: No discussion is provided regarding the capability of and requirements for using portable processing equipment for refueling outages.
C.III.1 11.3.1(2)	Design Bases	Conforms. Addressed in DCD Section 11.3.
C.III.1 11.3.1(3)	Design Bases	Conforms. Addressed in DCD Section 11.3.
C.III.1 11.3.1(4)	Design Bases	Conforms. Quality Assurance Program requirements are addressed in Chapter 17.
C.III.1 11.3.1(5)	Design Bases	Conforms. Addressed in DCD Section 11.3.5.
C.III.1 11.3.1(6)	Design Bases	Conforms. Addressed in DCD Section 12.6 and in Section 12.6.
C.III.1 11.3.1(7)	Design Bases	Conforms. Addressed in DCD Section 11.3.
C.III.1 11.3.2(1)	System Description	Conforms. Addressed in DCD Section 11.3.2.

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Section	Section Title	Conformance Evaluation
C.III.1 11.3.2(2)	System Description	Conforms. Addressed in DCD Section 11.3.2.
C.III.1 11.3.2(3)	System Description	Conforms. Addressed in DCD Section 11.3.2.
C.III.1 11.3.2(4)	System Description	Conforms. Addressed in DCD Sections 11.3.2, 11.3.3, and 9.4.
C.III.1 11.3.3	Radioactive Effluent Releases	Conforms. Addressed in DCD Sections 11.3 and 12.2, and in Section 12.2.
C.III.1 11.4.1(1)	Solid Waste Management System: Design Bases	Conforms. Addressed in DCD Section 11.4 and in Section 11.4.
C.III.1 11.4.1(2)	Design Bases	Conforms. Addressed in DCD Section 11.4 and in Section 11.4.
C.III.1 11.4.1(3)	Design Bases	Conforms. Addressed in DCD Section 11.4 and in Section 11.4.
C.III.1 11.4.1(4)	Design Bases	Conforms. Addressed in DCD Section 11.4 and in Sections 11.4, 13.5, and 17.5.
C.III.1 11.4.1(5)	Design Bases	Conforms. Addressed in DCD Section 11.4 and in Section 11.4.
C.III.1 11.4.1(6)	Design Bases	Conforms.
C.III.1 11.4.1(7)	Design Bases	Conforms. Addressed in DCD Section 11.4.
C.III.1 11.4.2(1)	System Description	Addressed in DCD Section 11.4 and in Section 11.4. Conforms with the following exception: The FSAR provides a description of the PCP. Detailed waste packaging methodologies will be provided in the PCP. The implementation milestone is provided in Section 13.4.
C.III.1 11.4.2(2)	System Description	Addressed in DCD Section 11.4 and in Section 11.4. Conforms with the following exception: The FSAR provides a description of the PCP. Detailed waste packaging methodologies will be provided in the PCP. The implementation milestone is provided in Section 13.4.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 11.4.2(3)	System Description	Addressed in DCD Section 11.4 and in Section 11.4. Conforms with the following exception: The FSAR provides a description of the PCP. Detailed waste packaging methodologies will be provided in the PCP. The implementation milestone is provided in Section 13.4. There are no temporary onsite storage facilities.
C.III.1 11.4.2 (4)	System Description	Conforms. Addressed in DCD Section 11.4.
C.III.1 11.4.3 (1)	Radioactive Effluent Releases	Addressed in DCD Section 11.4 and in Section 11.4. Conforms with the following exception: The FSAR provides a description of the PCP. Detailed waste packaging methodologies will be provided in the PCP. The implementation milestone is provided in Section 13.4.
C.III.1 11.4.3 (2)	Radioactive Effluent Releases	Conforms. Addressed in DCD Sections 3.1 and 11.4.
C.III.1 11.4.3 (3)	Radioactive Effluent Releases	Conforms. Addressed in DCD Section 12.2.
C.III.1 11.5.1	Process and Effluent Radiological Monitoring and Sampling Systems: Design Bases	Conforms
C.III.1 11.5.2(1)	System Description	Conforms. Addressed in DCD Section 11.5.
C.III.1 11.5.2 (2)	System Description	Conforms with the following exception: Section 11.5 provides a description of the ODCM. The implementation milestone is provided in Section 13.4.
C.III.1 11.5.2 (3)	System Description	Conforms with the following exception: Section 11.5 and TS Section 5 provide a description of radiological effluent controls. The implementation milestone is provided in Section 13.4.
C.III.1 11.5.2 (4)	System Description	Conforms with the following exception: Section 11.5 and TS Section 5 provide a description of the REMP. The implementation milestone is provided in Section 13.4.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 11.5.2 (5)	System Description	Conforms. Addressed in DCD Sections 3.1 and 11.5.
C.III.1 11.5.2 (6)	System Description	Conforms
C.III.1 11.5.2 (7)	System Description	Conforms
C.III.1 11.5.3	Effluent Monitoring and Sampling	Conforms
C.III.1 11.5.4	Process Monitoring and Sampling	Conforms
C.III.1 12.1.1	Policy Considerations	Conforms. Addressed in Sections 12.1 and 12.5.
C.III.1 12.1.2	Design Considerations	Conforms. Addressed in Section 12.5.
C.III.1 12.1.3	Operational Considerations	Conforms. Addressed in Sections 12.1 and 12.5.
C.III.1 12.2.1	Contained Sources	Conforms. Addressed in DCD Section 12.2.1.
C.III.1 12.2.2	Airborne Radioactive Material Sources	Conforms
C.III.1 12.3.1	Facility Design Features	Conforms
C.III.1 12.3.2	Shielding	Conforms
C.III.1 12.3.3	Ventilation	Conforms. Addressed in DCD Sections 9.4.1 and 12.3.
C.III.1 12.3.4	Area Radiation and Airborne Radioactivity Monitoring Instrumentation	Conforms. Addressed in Sections 12.3 and 12.5.
C.III.1 12.3.5	Dose Assessment	Conforms. Addressed in DCD Section 12.4 and in Section 12.4.
C.III.1 12.4	Dose Assessment	Conforms
C.III.1 12.5 (1) (a)	Operational Radiation Protection Program: Organization	Conforms. Addressed in Sections 12.5 and 13.1.
C.III.1 12.5 (1) (b)	Facilities	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 12.5 (1) (c)	Instrumentation and Equipment	Conforms
C.III.1 12.5 (1) (d)	Procedures	Conforms
C.III.1 12.5 (1) (e)	Training	Conforms. Addressed in Sections 12.5 and 13.2.
C.III.1 12.5 (2)		Conforms. Addressed in DCD Section 12.3.
C.III.1 12.5 (3)		Conforms. Addressed in Sections 12.5, 13.1, and 13.4.
C.III.1 12.5 (4)		Conforms. Addressed in Section 13.4.
C.III.1 12.5, last paragraph		Conforms. Addressed in Sections 12.5, 13.1, 13.2, and 13.5.
C.III.1 12.5.1	Organization	Conforms. Addressed in Sections 12.5 and 13.1.
C.III.1 12.5.2	Equipment, Instrumentation, and Facilities	Conforms
C.III.1 12.5.3	Procedures	Addressed in Sections 12.5, 13.2, 13.5, and 17.5. Conforms with one exception: With respect to RG 1.33, Dominion's QA procedures follow NQA-1 rather than the older standards referenced in RG 1.33. The QA requirements are described in Section 17.5.
C.III.1 13.1.1(1)	Organizational Structure of Applicant: Management and Technical Support Organization	Conforms. Addressed in Sections 13.1 and 14.2.
C.III.1 13.1.1(2)		Conforms
C.III.1 13.1.1(3)		Conforms
C.III.1 13.1.1(4)		Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 13.1.1(5)		Conforms
C.III.1 13.1.1(6)		Conforms
C.III.1 13.1.1(7)		Conforms. Addressed in Sections 13.1 and 14.2.
C.III.1 13.1.1.1	Design, Construction, and Operating Responsibilities	Conforms
C.III.1 13.1.1.2	Organizational Arrangement	Conforms. Addressed in Sections 13.1 and 17.5. Unit 3 is not a new, multi-unit plant site.
C.III.1 13.1.1.3	Qualifications	Conforms. Addressed in Sections 13.1 and 17.5.
C.III.1 13.1.2(1)		Exception. The guidelines of RG 1.33 are met through equivalent administrative controls described in Chapter 17.
C.III.1 13.1.2(2)		Exception. The guidelines of RG 1.33 are met through equivalent administrative controls described in Chapter 17.
C.III.1 13.1.2(3)		Conforms. Addressed in Sections 9.5.1 and 13.1.
C.III.1 13.1.2(4)		Conforms
C.III.1 13.1.2(5)		Conforms
C.III.1 13.1.2(6)		Conforms
C.III.1 13.1.2(7)		Conforms
C.III.1 13.1.2(8)		Conforms. Addressed in Appendix 13AA.
C.III.1 13.1.2.1	Plant Organization	Conforms. Addressed in Sections 13.1 and 17.5.
C.III.1 13.1.2.2(1)	Plant Personnel Responsibilities and Authorities	Conforms. Addressed in Sections 13.1 and 17.5.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 13.1.2.2(2)		Conforms
C.III.1 13.1.2.2(3)		Conforms
C.III.1 13.1.2.3	Operating Shift Crews	Conforms
C.III.1 13.1.3.1	Qualification Requirements	Conforms. Addressed in Sections 13.1 and 17.5.
C.III.1 13.1.3.2	Qualifications of Plant Personnel	Exception. Resumes will not be included in the application, but will be available for inspection at corporate headquarters upon request.
C.III.1 13.2.1	Plant Staff Training Program	Conforms
C.III.1 13.2.1.1 Licensed Staff (1)		Conforms with the following exceptions: 1) this item discusses inclusion of details of the licensed training program. As noted in Appendix 13BB, the systematic approach to training (SAT) process is used to establish and maintain training programs. Course duration and content are determined by the SAT process and by administrative procedure and are not included in the FSAR section; 2) the requirement for a “contingency plan…in the event fuel loading is subsequently delayed”' is met by the operator re-qualification program; and 3) the industry standard content for this section does not include a discussion of proposed schedule for licensed personnel.
C.III.1 13.2.1.1 Licensed Staff (2)		Conforms
C.III.1 13.2.1.1 Licensed Staff (3)		Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 13.2.1.1 Licensed Staff (4)		Conforms
C.III.1 13.2.1.1 Licensed Staff (5)		Conforms
C.III.1 13.2.1.1 Licensed Staff (6)		Conforms
C.III.1 13.2.1.1 Non-licensed Staff (1)		Conforms
C.III.1 13.2.1.1 Non-licensed Staff (2)		Conforms
C.III.1 13.2.1.1 Non-licensed Staff (3)		Exception – This item discusses programs not covered under 10 CFR 50.120. As noted in Appendix 13BB, the systematic approach to training (SAT) process is used to establish and maintain training programs. Course duration and content are determined by the SAT process and by administrative procedure and are not included in the FSAR section.
C.III.1 13.2.1.1 Non-licensed Staff (4)		Conforms. Addressed in Section 9.5.1.
C.III.1 13.2.1.1 Non-licensed Staff (5)		Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 13.2.1.1 Non-licensed Staff (6)		Conforms with the following exception: The first part of this item discusses detailed course descriptions. As noted in Appendix 13BB, the systematic approach to training (SAT) process is used to establish and maintain training programs. Course duration and content are determined by the SAT process and by administrative procedure and are not included in the FSAR section. The implementation milestone is addressed in Section 13.4.
C.III.1 13.2.1.1 Non-licensed Staff (7)		Conforms
C.III.1 13.2.1.2	Coordination with Preoperational Tests and Fuel Loading	Conforms with the following exception – Rather than providing contingency plans for training in the event of significantly delayed fuel loading the retraining programs are utilized, as described in Appendix 13BB. Figure 13.1-202 shows the training schedule relative to fuel loading.
C.III.1 13.2.2(1)	Applicable NRC Documents: 10 CFR 19	Conforms
C.III.1 13.2.2(2)	10 CFR 26	Conforms
C.III.1 13.2.2(3)	10 CFR 50	Conforms
C.III.1 13.2.2(4)	10 CFR 50 Appendix E	Conforms
C.III.1 13.2.2(5)	10 CFR 52	Conforms
C.III.1 13.2.2(6)	10 CFR 55	Conforms
C.III.1 13.2.2(7)	RG 1.8	Addressed in Table 1.9-202.
C.III.1 13.2.2(8)	RG 1.149	Addressed in Table 1.9-202.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 13.2.2(9)	NUREG-0711	Conforms. HFE addressed in DCD Chapter 18.
C.III.1 13.2.2(10)	NUREG-1021	Exception: Industry standard content for this section does not explicitly include discussion of compliance with NUREG-1021, Operator Licensing Examination Standards for Power Reactors.
C.III.1 13.2.2(11)	NUREG-1220	Not applicable. NUREG provides instructions for NRC inspectors.
C.III.1 13.2.2(12)	GL 86-04	Conforms
C.III.1 13.2.2(13)	RG 1.134	Conforms. Industry standard content for this section does not explicitly include a discussion of compliance with RG 1.134, Medical Evaluations.
C.III.1 13.3(1)	Emergency Planning	Conforms. Addressed in the Emergency Plan in COLA Part 5.
C.III.1 13.3(2)		Conforms. Addressed in the Emergency Plan in COLA Part 5.
C.III.1 13.3(3)		Conforms. Addressed in the Emergency Plan in COLA Part 5.
C.III.1 13.3(4)		Conforms. Addressed in Chapter 2, and the Emergency Plan and Evacuation Time Estimate in COLA Part 5.
C.III.1 13.3(5)		Conforms. Addressed in COLA Part 5.
C.III.1 13.3(6)		Not applicable. Applies when state and/or local governments decline to participate in emergency planning and preparedness.
C.III.1 13.3(7)		Conforms
C.III.1 13.3.1 (1)	Combined License Application and Emergency Plan Content	Conforms. Addressed in COLA Part 5.
C.III.1 13.3.1 (2)		Conforms. Addressed in COLA Part 5 and 10.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 13.3.1 (3)		Conforms. Addressed in Chapter 1 and the Emergency Plan in COLA Part 5.
C.III.1 13.3.1 (4)		Conforms. Addressed in the Emergency Plan in COLA Part 5.
C.III.1 13.3.1 (5)		Conforms. Addressed in the Emergency Plan in COLA Part 5.
C.III.1 13.3.1 (6)		Conforms. Addressed in the Emergency Plan in COLA Part 5.
C.III.1 13.3.1 (7)		Conforms. Addressed in Chapter 1.
C.III.1 13.3.1 (8)		Conforms. Addressed in the Emergency Plan in COLA Part 5.
C.III.1 13.3.1 (9)		Conforms. Addressed in the Emergency Plan in COLA Part 5.
C.III.1 13.3.2 (1)	Emergency Plan Considerations for Multiunit Sites	Conforms. The Unit 3 EP is a stand-alone plan and does not rely upon the EP for Units 1 and 2.
C.III.1 13.3.2 (2)		Not applicable. The Unit 3 EP is a stand-alone plan and does not rely upon the EP for Units 1 and 2.
C.III.1 13.3.2 (3)		Conforms. Addressed in the Emergency Plan in COLA Part 5 and 10.
C.III.1 13.3.2 (4)		Conforms. Addressed in COLA Part 5.
C.III.1 13.3.2 (5)		Conforms. Addressed in the Emergency Plan in COLA Part 5.
C.III.1 13.3.2 (6)		Conforms. Addressed in the Emergency Plan and the Evacuation Time Estimate in COLA Part 5.
C.III.1 13.3.2 (7)		Not applicable. Provisions for co-located licensees do not apply.
C.III.1 13.3.2 (8)		Conforms. Addressed in COLA Part 10.
C.III.1 13.3.2 (9)		Not applicable. There are no adjacent sites.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 13.3.3	Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria	Conforms with the following exceptions: 1. Did not include ITAAC in COLA Part 10 to address the non-bolded items in RG 1.206, Table II.C.1-B1. 2. Did not include ITAAC in COLA Part 10 to address RG 1.206, Table II.C.1-B1 ITAAC 17.0.
C.III.1 13.4	Operational Program Implementation	Conforms
C.III.1 13.5.1	Administrative Procedures	Conforms. Addressed in Sections 13.5 and 17.5.
C.III.1 13.5.2.1	Operating and Emergency Operating Procedures	Conforms with the following exception: Section 13.5.1 identifies classes of procedures by topic or type in lieu of the specific title. Operating procedures will be developed after activities such as job and task analyses have been completed.
C.III.1 13.5.2.2	Maintenance and Other Operating Procedures	Conforms
C.III.1 13.6	Security	Conforms. Addressed in Sections 13.4 and 13.6, and COLA Part 8.
C.I 13.7	FFD	Conforms
C.III.1 14.1	Verification Program: Specific Information to be Addressed for the Initial Plant Test Program	Conforms. Addressed in Sections 14.2 and 14.3.
C.III.1 14.2	Initial Plant Test Program	Conforms
C.III.1 14.2.1	Summary of Test Program and Objectives	Conforms
C.III.1 14.2.2	Organization and Staffing	Conforms. Addressed in DCD Section 14.2 and in Sections 13.1, 14.2, and 17.5.
C.III.1 14.2.3	Test Procedures	Conforms. Addressed in DCD Section 14.2.
C.III.1 14.2.4	Conduct of Test Program	Conforms. Addressed in DCD Section 14.2.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 14.2.5	Review, Evaluation, and Approval of Test Results	Conforms. Addressed in DCD Section 14.2.
C.III.1 14.2.6	Test Records	Conforms
C.III.1 14.2.7	Conformance of Tests Programs with Regulatory Guides	Conforms. Addressed in DCD Section 14.2.3.
C.III.1 14.2.8	Utilization of Reactor Operating and Testing Experiences in Development of Test Program	Conforms. Addressed in DCD Section 14.2 and in Section 14.2.
C.III.1 14.2.9	Trial Use of Plant Operating and Emergency Procedures	Conforms. Addressed in DCD Section 14.2.5 and in Section 13.2.
C.III.1 14.2.10	Initial Fuel Loading and Initial Criticality	Conforms. Addressed in DCD Section 14.2.6.
C.III.1 14.2.11	Test Program Schedule	Conforms. Addressed in DCD Section 14.2.7 and in Section 14.2.7.
C.III.1 14.2.12	Individual Test Descriptions	Conforms. Addressed in DCD Section 14.2.8 and in Section 14.2.9.
C.III.1 14.3	Inspections, Tests, Analyses, and Acceptance Criteria	Conforms. Addressed in COLA Part 10.
C.III.1 15.1	Transient and Accident Analyses: Transient and Accident Classification	Conforms. There are no aspects of the site-specific design that affect the transient and accident analyses in the DCD.
C.III.1 15.2	Frequency of Occurrence	Conforms
C.III.1 15.3	Plant Characteristics Considered in the Safety Evaluation	Conforms
C.III.1 15.4	Assumed Protection System Actions	Conforms
C.III.1 15.5	Evaluation of Individual Initiating Events	Conforms.
C.III.1 15.6	Event Evaluation	See below

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 15.6.1	Identification of Causes and Frequency Classification	Conforms
C.III.1 15.6.2	Sequence of Events and Systems Operation	Conforms
C.III.1 15.6.3	Core and System Performance	Conforms
C.III.1 15.6.4	Barrier Performance	Conforms
C.III.1 15.6.5	Radiological Consequences	Conforms. Table 2.0-201 compares the site-specific short-term χ/Qs for the EAB, LPZ, and control room to the χ/Qs assumed in the DCD.
C.III.1 16.1	Technical Specifications and Bases	Conforms. Addressed in COLA Part 4. There are no deviations from the generic TS bases.
C.III.1 16.2	Content and Format of Technical Specifications and Bases	Conforms. Addressed in COLA Part 4. No plant-specific deviations from the referenced certified generic Technical Specifications or Bases are required and none are being requested (e.g., incorporation of TSTF travelers).
C.III.1 17.1	Quality Assurance and Reliability Assurance: Quality Assurance During the Design and Construction Phase	Conforms
C.III.1 17.2	Quality Assurance During the Operations Phase	Conforms
C.III.1 17.3	Quality Assurance Program Description	Conforms
C.III.1 17.4.1	New Section 17.4 in the Standard Review Plan	Conforms
C.III.1 17.4.2	Reliability Assurance Program Scope, Stages, and Goals	Not applicable
C.III.1 17.4.3	Reliability Assurance Program Implementation	Conforms. Addressed in Sections 17.4 and 17.6.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 17.4.4	Reliability Assurance Program Information Needed in a COL Application	Conforms. Addressed in DCD Section 17.4 and in Sections 17.4, 17.5, and 17.6.
C.III.1 17.5	Quality Assurance Program Guidance	See below
C.III.1 17.5.1	COL Applicant QA Program Responsibilities	Conforms
C.III.1 17.5.2	Updated SRP Section 17.5 and the QA Program Description	Conforms. QA applied to safety-related activities performed prior to the start of construction (e.g., site investigation, design and safety analysis, early procurements) is described in the Dominion Nuclear Facility QAPD topical report, DOM-QA-1. QA applied during activities to adapt the design to specific plant implementation, construction, and operations is addressed in Section 17.5.
C.III.1 17.5.3	Evaluation of the QAPD Against the SRP and QAPD Submittal Guidance	Conforms
C.III.1 17.6	Description of the Applicant's Program for Implementation of 10 CFR 50.65, the Maintenance Rule	Conforms
C.III.1 17.6.1	Scoping per 10 CFR 50.65(b)	Conforms
C.III.1 17.6.2	Monitoring per 10 CFR 50.65(a)	Conforms
C.III.1 17.6.3	Periodic Evaluation per 10 CFR 50.65(a)(3)	Conforms
C.III.1 17.6.4	Risk Assessment and Management per 10 CFR 50.65(a)(4)	Conforms
C.III.1 17.6.5	Maintenance Rule Training and Qualification	Conforms

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.III.1 17.6.6	Maintenance Rule Program Role in Implementation of Reliability Assurance Program (RAP) in the Operations Phase	Conforms
C.III.1 17.6.7	Maintenance Rule Program Implementation	Conforms
C.III.1 Chapter 18	Human Factors Engineering	Conforms
	HFE principles incorporated into:	
	(1) Planning and management	Conforms. Addressed in DCD Section 18.2.
	(2) Plant design processes not closed with design certification	Conforms. Addressed in DCD Tier 1, ITAAC Table 3.3-1.
	(3) HSI, procedures, and training	Conforms. Addressed in DCD Tier 1, ITAAC Table 3.3-1, Items 6, 7, and 8.
	(4) implementation of the design	Conforms. Addressed in DCD Tier 1, ITAAC Table 3.3-1, Item 10.
	(5) monitoring of performance at the site	Conforms. Addressed in DCD Tier 1, ITAAC Table 3.3-1, Item 11.
	Applicant program addresses normal and emergency, maintenance, test, inspection and surveillance activities	Conforms. Addressed in DCD Section 18.1.
	FSAR/DCD describe objectives and scope of the applicant's activities related to element, methodology, and results for (12 HFE elements)	Conforms. Addressed in DCD Sections 18.3 through 18.13.
	Applicant should reference detailed implementation plan reviewed and approved as part of design certification	Conforms. Addressed in DCD Section 18.2.1.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.I 18.1	HFE Program Management	Conforms. Addressed in DCD Sections 18.2.2 and 18.2.3.
C.I 18.1.1	General HFE Program and Scope	Conforms. Addressed in DCD Sections 18.2.1 and 18.2.2.
C.I 18.1.2	HFE Team and Organization	Conforms. Addressed in DCD Section 18.2.3.
C.I 18.1.3	HFE Process and Procedures	Conforms. Addressed in DCD Sections 18.2.1 and 18.2.2.
C.I 18.1.4	HFE Issues Tracking	Conforms. Addressed in DCD Section 18.2.2.
C.I 18.1.5	HFE Technical Program	Conforms. Addressed in DCD Sections 18.3 through 18.13.
C.I 18.2.1	Objectives and scope	Conforms. Addressed in DCD Section 18.3.1.
C.I 18.2.2.1	OER Process	Conforms. Addressed in DCD Section 18.3.2.
C.I 18.2.2.2	Predecessor plants and systems	Conforms. Addressed in DCD Section 18.3.2.1.
C.I 18.2.2.3	Risk-important human actions	Conforms. Addressed in DCD Section 18.3.2.2.
C.I 18.2.2.4	HFE technology	Conforms. Addressed in DCD Section 18.3.2.3.
C.I 18.2.2.5	Recognized industry issues	Conforms. Addressed in DCD Section 18.3.2.4.
C.I 18.2.2.6	Issued Identified by plant personnel	Conforms. Addressed in DCD Section 18.3.2.5.
C.I 18.2.2.7	Issue Analysis, Tracking, and Review	Conforms. Addressed in DCD Section 18.3.2.6.
C.I 18.2.3	Results	Conforms. Addressed in DCD Section 18.3.3.
C.I 18.3.1	Objectives and Scope	Conforms. Addressed in DCD Section 18.4.2.
C.I 18.3.1.1	Functional Requirements Analysis	Conforms. Addressed in DCD Section 18.4.1.
C.I 18.3.1.2	Function Allocation Analysis	Conforms. Addressed in DCD Section 18.4.2.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.I 18.3.2.1	Methodology for Functional Requirements Analysis	Conforms. Addressed in DCD Section 18.4.1.
C.I 18.3.2.2	Methodology for Function Allocation Analysis	Conforms. Addressed in DCD Section 18.4.2.
C.I 18.3.3	Results	Conforms. Addressed in DCD Sections 18.4.1 and 18.4.2.
C.I 18.4.1	Objectives and Scope	Conforms. Addressed in DCD Section 18.5.1.
C.I 18.4.2	Methodology	Conforms. Addressed in DCD Section 18.5.1.
C.I 18.4.3	Results	Conforms. Addressed in DCD Section 18.5.1.
C.I 18.5.1	Objectives and Scope	Conforms. Addressed in DCD Section 18.6.2.
C.I 18.5.2	Methodology	Conforms. Addressed in DCD Sections 18.6.4 and 18.6.5.
C.I 18.5.3	Results	Conforms. Addressed in DCD Section 18.6.6.
C.I 18.6.1	Objectives and Scope	Conforms. Addressed in DCD Section 18.7.1.
C.I 18.6.2	Methodology	Conforms. Addressed in DCD Section 18.7.2.
C.I 18.6.3	Results	Conforms. Addressed in DCD Section 18.7.3.
C.I 6.3.2.8	Manual Actions	Conforms. Addressed in DCD Section 18.7.2.
C.I 18.7.1	Objectives and scope	Conforms. Addressed in DCD Section 18.8.1.
C.I 18.7.2.1	HSI Design Inputs	Conforms. Addressed in DCD Section 18.8.1.
C.I 18.7.2.2	Concept of operations	Conforms. Addressed in DCD Section 18.8.1.
C.I 18.7.2.3	Functional Requirements Specification	Conforms. Addressed in DCD Section 18.8.1.
C.I 18.7.2.4	HSI Concept Design	Conforms. Addressed in DCD Section 18.8.1.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.I 18.7.2.5	HSI Detailed Design and Integration	Conforms. Addressed in DCD Section 18.8.1.
C.I 18.7.2.6	HSI Tests and Evaluations	Conforms. Addressed in DCD Section 18.8.1.
C.I 18.7.3.1	Overview of HSI Design and Its Key Features	Conforms. Addressed in DCD Section 18.8.1(3).
C.I 18.7.3.2	Safety Aspects of the HSI	Conforms. Addressed in DCD Section 18.8.1(3).
C.I 18.7.3.3	HSI Change Process	Conforms. Addressed in DCD Section 18.13.3.
C.I 18.8.1	Objectives and Scope	Conforms. Addressed in DCD Section 18.9.1.
C.I 18.8.2	Methodology	Conforms. Addressed in DCD Section 18.9.2.
C.I 18.8.3	Results	Conforms. Addressed in DCD Section 18.9.3.
C.I 18.9.1	Objectives and Scope	Conforms. Addressed in DCD Sections 18.10.1 and 18.10.2.
C.I 18.9.2	Methodology	Conforms. Addressed in DCD Sections 18.10.3 and 18.10.4.
C.I 18.9.3	Results	Conforms. Addressed in DCD Section 18.10.5.
C.I 18.10.1	Objectives and Scope	Conforms. Addressed in DCD Section 18.11 and 18.11.1.
C.I 18.10.2	Methodology	Conforms. Addressed in DCD Section 18.11.
C.I 18.10.2.1	Operational Conditions Sampling	Conforms. Addressed in DCD Section 18.11.
C.I 18.10.2.2	Design Verification	Conforms. Addressed in DCD Section 18.11.
C.I 18.10.2.3	Integrated System Validation	Conforms. Addressed in DCD Section 18.11.
C.I 18.10.2.4	Human Engineering Discrepancy Resolution	Conforms. Addressed in DCD Section 18.11.
C.I 18.10.3	Results	Conforms. Addressed in DCD Section 18.11.2.
C.I 18.11.1	Objectives and Scope	Conforms. Addressed in DCD Section 18.12.1.

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Table 1.9-203 Conformance With the FSAR Content Guidance In RG 1.206

Section	Section Title	Conformance Evaluation
C.I 18.11.2	Methodology	Conforms. Addressed in DCD Section 18.12.2.
C.I 18.11.3	Results	Conforms. Addressed in DCD Section 18.12.3.
C.I 18.12.1	Objectives and Scope	Conforms. Addressed in DCD Sections 18.13.1 and 18.13.2.
C.I 18.12.2	Methodology	Conforms. Addressed in DCD Sections 18.13.2 and 18.13.3.
C.I 18.12.3	Results	Conforms. Addressed in DCD Section 18.13.4.
C.III.1 Chapter 19	Probabilistic Risk Assessment and Severe Accident Evaluation	Conforms. As discussed in RG 1.206, Section C.III.1.10, the FSAR follows the organization and numbering of the referenced certified design.

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Table 1.9-204 Industrial Codes and Standards

Code or Standard Number	Year	Title
American National Standards Institute		
N323D	2002	Installed Radiation Protection Instrumentation
American Society of Civil Engineers (ASCE)		
ASCE 7-02	2002	Minimum Design Loads for Buildings and Other Structures
American Society of Mechanical Engineers (ASME)		
A17.1	2007	Safety Code for Elevators and Escalators
B31.1	2007	Power Piping
NQA-1	1994	Quality Assurance Requirements for Nuclear Facility Applications
Boiler and Pressure Vessel Code, Section IX	2007	Qualification Standard for Welding and Brazing Procedures, Welder, Brazers and Welding and Brazing Operators
ASTM International		
ASTM E84-07	2007	Standard Test Method for Surface Burning Characteristics of Building Materials
ASTM E119-07a	2007	Standard Test Methods for Fire Tests of Building Construction and Materials
ASTM E814-06	2006	Standard Test Method for Fire Tests of Through-Penetration Fire Stops
Applicable Building Codes		
International Building Code	As defined in the Virginia Uniform Statewide Building Code edition of record	International Building Code
International Fire Code	As defined in the Virginia Uniform Statewide Building Code edition of record	International Fire Code

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Table 1.9-204 Industrial Codes and Standards

Code or Standard Number	Year	Title
Applicable Building Codes (continued)		
28 CFR 36		Nondiscrimination on the Basis of Disability by Public Accommodations and in Commercial Facilities (Americans With Disabilities Act (ADA) Accessibility Guidelines)
	2003	Virginia Uniform Statewide Building Code, Part I (Virginia Construction Code)
Factory Mutual		
Data Sheet 7-42	2006	Guidelines for Evaluating the Effects of Vapor Cloud Explosions Using a TNT Equivalency Method
	2007	Approval Guide
Institute of Electrical and Electronics Engineers (IEEE)		
C2	2007	National Electric Safety Code
C57.19.100-1995 (R2003)	2004	IEEE Guide for Application of Power Apparatus Bushings
National Fire Protection Association (NFPA)		
NFPA 10	2007	Standard for Portable Fire Extinguishers
NFPA 11	2005	Standard for Low-, Medium-, and High-Expansion Foam
NFPA 13	2007	Standard for the Installation of Sprinkler Systems
NFPA 14	2007	Standard for the Installation of Sandpipe and Hose Systems
NFPA 15	2007	Standard for Water Spray Fixed Systems for Fire Protection
NFPA 16	2007	Standard for the Installation of Foam-Water Sprinkler and Foam-Water Spray Systems
NFPA 20	2007	Standard for the Installation of Stationary Pumps for Fire Protection
NFPA 24	2007	Standard for the Installation of Private Fire Service Mains and their Appurtenances

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Table 1.9-204 Industrial Codes and Standards

Code or Standard Number	Year	Title
		NFPA (continued)
NFPA 25	2008	Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems
NFPA 30	2008	Flammable and Combustible Liquids Code
NFPA 37	2006	Standard for the Installation and Use of Stationary Combustion Engines and Gas Turbines
NFPA 55	2005	Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks
NFPA 70	2008	National Electric Code
NFPA 72	2007	National Fire Alarm Code
NFPA 80	2007	Standard for Fire Doors and Other Opening Protectives
NFPA 80A	2007	Recommended Practice for Protection of Buildings from Exterior Fire Exposures
NFPA 101	2006	Life Safety Code
NFPA 204	2007	Standard for Smoke and Heat Venting
NFPA 214	2005	Standard on Water-Cooling Towers
NFPA 241	2004	Standard for Safeguarding Construction, Alteration, and Demolition Operations
NFPA 252	2008	Standard Methods of Fire Tests of Door Assemblies
NFPA 255	2006	Standard Method of Test of Surface Burning Characteristics of Building Materials
NFPA 780	2008	Standard for the Installation of Lightning Protection Systems

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Table 1.9-204 Industrial Codes and Standards

Code or Standard Number	Year	Title
North American Electric Reliability Corporation (NERC)		
PRC-005-1	2006	Transmission and Generation Protection System Maintenance and Testing
PRC-008-0	2005	Underfrequency Load Shedding Equipment Maintenance Program
PRC-017-0	2005	Special Protection System Maintenance and Testing
Occupational Safety and Health Act (OSHA)		
29 CFR 1910	2006	Occupational Safety and Health Standards
29 CFR 1926	2006	Safety and Health Regulations for Construction
Underwriters Laboratories (UL)		
	2007	Fire Protection Equipment Directory
Environmental Protection Agency (EPA)		
40 CFR 60	2006	EPA Standards of Performance for Stationary Compression Ignition Internal Combustion Engines

NAPS SUP 1.9-2

Table 1.9-205 NUREG Reports Cited

NUREG No.	Issue Date	Title	Comment/ Section Where Discussed
0016, Rev. 1	01/1979	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWRs)	12.2
0570	06/1979	Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release	6.4
0612	07/1980	Control of Heavy Loads at Nuclear Power Plants	13.5
0737	11/1980	Clarification of TMI Action Plan Requirements	13.1
0800	03/2007	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants	1.1 2.0 2.2 2.5 9.3 11.5
0868	06/1982	A Collection of Mathematical Models for Dispersion in Surface Water and Groundwater	2.4
1437	05/1996	Generic Environmental Impact Statement for License Renewal of Nuclear Plants	12.2
1736	10/2001	Consolidated Guidance: 10 CFR Part 20 – Standards for Protection Against Radiation	1.9
1805	12/2004	Fire Dynamics Tools (FDT^{s}) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program	2.2
1811, Vol. 1	12/2006	Environmental Impact Statement for an Early Site Permit (ESP) at the North Anna ESP Site, Volume 1	2.4
1835	09/2005	Safety Evaluation Report for an Early Site Permit (ESP) at the North Anna ESP Site	2.0

NAPS SUP 1.9-2

Table 1.9-205 NUREG Reports Cited

NUREG No.	Issue Date	Title	Comment/ Section Where Discussed
CR-4013	04/1986	LADTAP II Technical Reference and User Guide	12.2
CR-4653	03/1987	GASPAR II Technical Reference and User Guide	12.2
CR-5512, Vol. 1	10/1992	Residual Radioactive Contamination from Decommissioning, Vol. 1	2.4
CR-6624	11/1999	Recommendations for Revision of Regulatory Guide 1.78	2.2
CR-6697	11/2000	Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes	2.4
CR-6728	10/2001	Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines	2.5

1.10 Summary of COL Items

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

Add the following at the end of this section.

NAPS SUP 1.10-1

Table 1.10-201 lists the FSAR location(s) where the individual COL items from the DCD are addressed. Table 1.10-202 lists the FSAR location(s) where the individual COL Action Items and Permit Conditions from the ESP (Reference 1.10-202) are addressed.

1.10 References

1.10-201 [Deleted]

1.10-202 Early Site Permit (ESP) for the North Anna ESP Site, No. ESP-003, U.S. Nuclear Regulatory Commission, November 2007.

NAPS SUP 1.10-1

Table 1.10-201 Summary of FSAR Sections Where DCD COL Items Are Addressed

Item No.	Subject/Description of Item	FSAR Section
1.1-1-A	Establish Rated Electrical Output	1.1.2.7
1.3-1-A	Update Table 1.3-1	1.3.1
1.7-1-H	Final Design Configuration Confirmation	1.7
1.9-3-A	SRP and Regulatory Guide Applicability	SRP: Table 1.9-201 RGs: 1.9.1 and 1.9.2 RG 1.206: Table 1.9-203
1.11-1-A	Address Table 1.11-1 Items that refer to Notes (2) and (7)	1.11.1 and Table 1.11-201
1C.1-1-A	Handling of Safeguards Information	1C.1, Table 1C-201
1C.1-2-A	Emergency Preparedness and Response Actions	1C.1, Table 1C-202
2.0-1-A	Site Characteristics Demonstration	2.0
2.0-2-A	Site Location and Description Information in Accordance with SRP 2.1.1	2.0 and 2.1.1
2.0-3-A	Site-Specific Exclusion Area Authority and Control Information in Accordance with SRP 2.1.2.	2.0 and 2.1.2
2.0-4-A	Describe the Population Distribution in Accordance with SRP 2.1.3	2.0 and 2.1.3
2.0-5-A	Identify Potential Hazards in the Site Vicinity, in Accordance with SRP 2.2.1 - 2.2.2	2.0 and 2.2
2.0-6-A	Evaluation of Potential Accidents in Accordance with SRP 2.2.3	2.0 and 2.2.3
2.0-7-A	Regional Climatology in Accordance with SRP 2.3.1	2.0 and 2.3.1
2.0-8-A	Local Meteorology in Accordance with SRP 2.3.2	2.0 and 2.3.2
2.0-9-A	Onsite Meteorological Measurement Programs in Accordance with SRP 2.3.3	2.0 and 2.3.3
2.0-10-A	Short-Term Diffusion Estimates for Accidental Atmospheric Releases in Accordance with SRP 2.3.4	2.0 and 2.3.4

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Table 1.10-201 Summary of FSAR Sections Where DCD COL Items Are Addressed

Item No.	Subject/Description of Item	FSAR Section
2.0-11-A	Long-Term Diffusion Estimates in Accordance with SRP 2.3.5	2.0 and 2.3.5
2.0-12-A	Hydraulic Description Maximum Ground Water Level in Accordance with SRP 2.4.1	2.0 and 2.4.1
2.0-13-A	Protection of Below-Grade Penetrations and Access Openings from Floods in Accordance with SRP 2.4.2	2.0 and 2.4.2
2.0-14-A	Probable Maximum Flood on Streams and Rivers in Accordance with SRP 2.4.3	2.0 and 2.4.3
2.0-15-A	Potential Dam Failures Seismically Induced in Accordance with SRP 2.4.4	2.0 and 2.4.4
2.0-16-A	Probable Maximum Surge and Seiche Flooding in Accordance with SRP 2.4.5	2.0 and 2.4.5
2.0-17-A	Probable Maximum Tsunami in Accordance with SRP 2.4.6	2.0 and 2.4.6
2.0-18-A	Ice Effects in Accordance with SRP 2.4.7	2.0 and 2.4.7
2.0-19-A	Cooling Water Canals and Reservoirs in Accordance with SRP 2.4.8	2.0 and 2.4.8
2.0-20-A	Channel Diversion in Accordance with SRP 2.4.9	2.0 and 2.4.9
2.0-21-A	Flooding Protection Requirements in Accordance with SRP 2.4.10	2.0 and 2.4.10
2.0-22-A	Cooling Water Supply in Accordance with SRP 2.4.11	2.0 and 2.4.11
2.0-23-A	Groundwater in Accordance with SRP 2.4.12	2.0 and 2.4.12
2.0-24-A	Accidental Releases of Liquid Effluents in Ground and Surface Waters in Accordance with SRP 2.4.13	2.0 and 2.4.13
2.0-25-A	Technical Specifications and Emergency Operation Requirements in Accordance with SRP 2.4.14	2.0 and 2.4.14
2.0-26-A	Basic Geologic and Seismic Information in Accordance with SRP 2.5.1	2.0 and 2.5.1
2.0-27-A	Vibratory Ground Motion in Accordance with SRP 2.5.2	2.0 and 2.5.2

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Table 1.10-201 Summary of FSAR Sections Where DCD COL Items Are Addressed

Item No.	Subject/Description of Item	FSAR Section
2.0-28-A	Surface Faulting in Accordance with SRP 2.5.3	2.0 and 2.5.3
2.0-29-A	Stability of Subsurface Materials and Foundations in Accordance with SRP 2.5.4	2.0 and 2.5.4
2.0-30-A	Stability of Slopes in Accordance with SRP 2.5.5	2.0 and 2.5.5
2A.2-1-A	Confirmation of the ESBWR χ/Q Values	2.3.4.3 and 2A.2.4
2A.2-2-A	Confirmation of the Reactor Building χ/Q Values	2A.2.5
3.9.9-1-H	Reactor Internals Vibration Analysis, Measurement and Inspection Program	3.9.2.4
3.9.9-2-H	ASME Class 2 or 3 or Quality Group D Components with 60 Year Design Life	3.9.3.1
3.9.9-3-A	Inservice Testing Programs	3.9.6
3.9.9-4-A	Snubber Inspection and Test Program	3.9.3.7.1(3)e
3.10.4-1-A	Dynamic Qualification Report	3.10.1.4
3.11-1-A	Environmental Qualification Document (EQD)	3.11.4.4
4.3-1-A	Variances from Certified Design	4.3.3.1
4A-1-A	Variances from Certified Design	4A.1
5.2-1-A	Preservice and Inservice Inspection Program Description	5.2.4, 5.2.4.3.4, 5.2.4.6, 5.2.4.11, and 6.6
5.2-2-H	Leak Detection Monitoring	5.2.5 and 5.2.5.9
5.2-3-A	Preservice and Inservice Inspection NDE Accessibility Plan Description	5.2.4 and 5.2.4.2
5.3-2-A	Materials and Surveillance Capsule	5.3.1.8
6.2-1-H	Pipe Length from Containment to Inboard/Outboard Isolation Valve	6.2.4.2
6.4-1-A	CRHA Procedures and Training	6.4.4
6.4-2-A	Toxic Gas Analysis	6.4.5
6.6-1-A	PSI/ISI Program Description	6.6
6.6-2-A	PSI/ISI NDE Accessibility Plan Description	6.6.2

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Table 1.10-201 Summary of FSAR Sections Where DCD COL Items Are Addressed

Item No.	Subject/Description of Item	FSAR Section
8.2.4-1-A	Transmission System Description	8.2.1.1
8.2.4-2-A	Switchyard Description	8.2.1.2.1
8.2.4-3-A	Normal Preferred Power	8.2.1.2
8.2.4-4-A	Alternate Preferred Power	8.2.1.2
8.2.4-5-A	Protective Relaying	8.2.1.2.2
8.2.4-6-A	Switchyard DC Power	8.2.1.2.1
8.2.4-7-A	Switchyard AC Power	8.2.1.2.1
8.2.4-8-A	Switchyard Transformer Protection	8.2.1.2.1
8.2.4-9-A	Stability and Reliability of the Offsite Transmission Power Systems	8.2.2.1
8.2.4-10-A	Interface Requirements	8.2.2.1
8A.2.3-1-A	Cathodic Protection System	8A.2.1
9.1-4-A	Fuel Handling Operations	9.1.4.13 and 9.1.4.19
9.1-5-A	Handling of Heavy Loads	9.1.5.6, 9.1.5.8, and 9.1.5.9
9.2.1-1-A	Material Selection	9.2.1.2
9.2.5-1-H	Post 7-Day Makeup to Ultimate Heat Sink (UHS)	9.2.5
9.3.2-1-A	Post-Accident Sampling Program	9.3.2.2
9.3.9-1-A	Implementation of Hydrogen Water Chemistry	9.3.9
9.3.9-2-A	Hydrogen and Oxygen Storage and Supply	9.3.9.2
9.3.10-1-A	Oxygen Storage Facility	9.3.10.2
9.3.11-1-A	Determine Need for Zinc Injection System	9.3.11.2
9.3.11-2-A	Provide System Description for Zinc Injection System	9.3.11.4
9.5.1-1-A	Secondary Firewater Storage Source	9.5.1.4
9.5.1-2-A	Secondary Firewater Capacity	9.5.1.4
9.5.1-4-A	Piping and Instrument Diagrams	9.5.1.2, 9.5.1.4, 9.5.1.5, and Figures 9.5-201, 9.5-202, and 9.5-203

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Table 1.10-201 Summary of FSAR Sections Where DCD COL Items Are Addressed

Item No.	Subject/Description of Item	FSAR Section
9.5.1-5-A	Fire Barriers	9.5.1.10
9.5.1-6-H	Smoke Control	9.5.1.11
9.5.1-7-H	Fire Hazards Analysis (FHA) Compliance Review	9.5.1.12
9.5.1-8-A	Fire Protection (FP) Program Description	9.5.1.15
9.5.1-10-H	Fire Brigade	9.5.1.15.4, 13.1.2.1.5
9.5.1-11-A	Quality Assurance	9.5.1.15.9
9.5.2.5-1-A	Emergency Notification System	9.5.2.2
9.5.2.5-2-A	Grid Transmission Operator	9.5.2.2
9.5.2.5-3-A	Offsite Interfaces (1)	9.5.2.2
9.5.2.5-4-A	Offsite Interfaces (2)	9.5.2.2
9.5.2.5-5-A	Fire Brigade Radio System	9.5.2.2
9.5.4-1-A	Fuel Oil Capacity	9.5.4.2
9.5.4-2-A	Protection of Underground Piping	9.5.4.2
9A.7-1-A	Yard Fire Zone Drawings	9A.4.7
9A.7-2-A	Fire Hazards Analysis for Site Specific Areas	9A.4.7, 9A.5.7, 9A.5.8, 9A.5.9, and 9A.5.12
10.2-1-A	Turbine Maintenance and Inspection Program	10.2.3.6
10.2-2-A	Turbine Missile Probability Analysis	10.2.3.8
10.4-1-A	Leakage (of Circulating Water Into the Condenser)	10.4.6.3
11.2-1-A	Implementation of IE Bulletin 80-10	11.2.2.3
11.2-2-A	Implementation of Part 20.1406	11.2.2.3
11.4-1-A	SWMS Processing Subsystem Regulatory Guide Compliance	11.4.2.3.5
11.4-2-A	Compliance with IE Bulletin 80-10	11.4.2.3.5
11.4-3-A	Process Control Program	11.4.2.3.5
11.4-4-A	Temporary Storage Facility	11.4.1
11.4-5-A	Compliance with Part 20.1406	11.4.1

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Table 1.10-201 Summary of FSAR Sections Where DCD COL Items Are Addressed

Item No.	Subject/Description of Item	FSAR Section
11.5-1-A	Sensitivity or Subsystem Lower Limit of Detection	11.5.4.7
11.5-2-A	Offsite Dose Calculation Manual	11.5.4.4, 11.5.4.5, and 11.5.5.8
11.5-3-A	Process and Effluent Monitoring Program	11.5, 11.5.4.6, and Table 11.5-201
11.5-4-A	Site Specific Offsite Dose Calculation	11.5.4.8
11.5-5-A	Instrument Sensitivities	11.5.4.9
12.1-1-A	Regulatory Guide 8.10	12BB
12.1-2-A	Regulatory Guide 1.8	12BB
12.1-3-A	Operational Considerations	12BB
12.1-4-A	Regulatory Guide 8.8	12BB
12.2-2-A	Airborne Effluents and Doses	12.2.2.1, 12.2.2.2, and Table 2.0-201
12.2-3-A	Liquid Effluents and Doses	12.2.2.4
12.2-4-A	Other Contained Sources	12.2.1.5
12.3-2-A	Operational Considerations	12.3.4
12.3-3-H	Controlled Access	12.3.1.3
12.5-1-A	Equipment, Instrumentation, and Facilities	12BB
12.5-2-A	Compliance with 10 CFR Part 50.34(f)(2)(xxvii) and NUREG-0737 Item III.D.3.3	12BB
12.5-3-A	Radiation Protection Program	12BB
13.1-1-A	Organizational Structure	9.5.1.15.3, 13.1.1 through 13.1.3, and Appendix 13AA
13.2-1-A	Reactor Operator Training	13.2.1 and 13BB
13.2-2-A	Training for Non-Licensed Plant Staff	13.2.2 and 13BB
13.3-1-A	Identification of OSC and Communication Interfaces with Control Room and TSC	13.3 and COLA Part 5, Sections II.F and II.H
13.3-2-A	Identification of EOF and Communication Interfaces with Control Room and TSC	13.3 and COLA Part 5, Sections II.F and II.H

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Table 1.10-201 Summary of FSAR Sections Where DCD COL Items Are Addressed

Item No.	Subject/Description of Item	FSAR Section
13.3-3-A	Decontamination Facilities	13.3 and COLA Part 5, Section II.J
13.4-1-A	Operation Programs	13.4
13.4-2-A	Implementation Milestones	13.4
13.5-1-A	Administrative Procedures Development Plan	13.5.1
13.5-2-A	Plant Operating Procedures Development Plan	13.5.2
13.5-3-A	Emergency Procedures Development	13.5.2
13.5-5-A	Procedures Included in Scope of Plan	13.5, 13.5.2
13.5-5-A	Procedures Included in Scope of Plan	13.5.2
13.5-6-H	Procedures for Calibration, Inspection, and Testing	13.5.2
13.6-6-A	Key Control	13.6.1.1.5
13.6-7-A	Secondary Alarm Station Design	Physical Security Plan
13.6-8-H	CAS and SAS Redundancy	13.6.1.1.8
13.6-9-A	Operational Alarm Response Procedures	13.6.1.1.3
13.6-10-A	Operational Surveillance Test Procedures	13.6.1.1.8
13.6-11-A	Maintenance Test Procedures	13.6.1.1.8
13.6-12-A	Operational Response Procedures to Security Events	13.6.2
13.6-13-A	Operational Alarm Response Procedures	13.6.1.1.3
13.6-14-A	Administrative Controls to Sensitive Cabinets	13.6.1.1.5
13.6-15-A	Administrative Controls to Sensitive Equipment	13.6.1.1.5
14.2-1-A	Description - Initial Test Program Administration	14.2.2.1, Appendix 14AA
14.2-2-H	Startup Administrative Manual	14.2.2.1
14.2-3-H	Test Procedures	14.2.2.2
14.2-4-H	Test Program Schedule and Sequence	14.2.7
14.2-5-A	Site Specific Tests	14.2.9

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Table 1.10-201 Summary of FSAR Sections Where DCD COL Items Are Addressed

Item No.	Subject/Description of Item	FSAR Section
14.2-6-H	Site Specific Test Procedures	14.2.9
14.3-1-A	Emergency Planning ITAAC	14.3.8
14.3-2-A	Site-Specific ITAAC	14.3.9
14.3A-1-1	Establish a Schedule for Design Acceptance Criteria ITAAC Closure	14.3A.1
16.0-1-A	COL Applicant Bracketed Items	COLA Part 4
16.0-2-H	COL Holder Bracketed Items	5.3.1.5, COLA Part 4
17.2-1-A	QA Program for the Construction and Operations Phases	17.2
17.2-2-A	QA Program for Design Activities	17.2
17.3-1-A	Quality Assurance Program Document	17.3
17.4-1-H	Operation Reliability Assurance Activities	17.4.1, 17.4.6, 17.4.9, 17.4.10, and 17.6
18.13-1-H	Milestone for HPM Implementation	18.13.3
19.2.6-1-H	Seismic High Confidence Low Probability of Failure Margins	19.2.3.2.4

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Table 1.10-202 Summary of FSAR Sections Where ESP COL Action Items and Permit Conditions Are Addressed

Item No.	Subject/Description of Item	Section
ESP 2.1-1	Provide Latitude, Longitude, and UTM Coordinates	2.1.1
ESP 2.1-2	Control of Lake in Exclusion Area	2.1.2
ESP 2.2-1	Evaluate Industrial Hazards Near the Site	2.2
ESP 2.2-2	Interactions between Existing Units	2.2
ESP 2.3-1	Cooling Towers Impacts	2.3
ESP 2.3-2	Dispersion to Control Room	2.3
ESP 2.3-3	Verify Long-Term Atmospheric Dispersion Characteristics	2.3
ESP 2.4-1	Layout of Intake and Discharge Tunnels (Plant Service Water and Circulating Water System)	1.12
ESP 2.4-2	Plant Shutdown Protocol for Minimum Lake Level	2.4.14
ESP 2.4-4	Grading for Drainage	2.4.2
ESP 2.4-5	Local Probable Maximum Precipitation (PMP) Flooding Protection Needs	2.4.2
ESP 2.4-6	Engineered Underground Ultimate Heat Sink (UHS) Design	2.4.4
ESP 2.4-7	Engineered Underground UHS Capacity	2.4.4
ESP 2.4-8	Address Safety-Related Withdrawals from Lake	2.4.8
ESP 2.4-9	Slope Embankment Protection for Intake Structure	2.4.10
ESP 2.4-10	Cooling Water Needs at Low Lake Levels	2.4.11
ESP 2.5-1	Perform Additional Borings	2.5.1
ESP 2.5-2	Plot Plans and Profiles	2.5.4
ESP 2.5-3	Provide Excavation and Backfill Plans	2.5.4
ESP 2.5-4	Groundwater Conditions	2.5.4
ESP 2.5-5	Perform Additional Soil Column Amplification and Attenuation Analyses	2.5.4
ESP 2.5-6	Safety-Related Facilities Stability Analysis	2.5.4

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Table 1.10-202 Summary of FSAR Sections Where ESP COL Action Items and Permit Conditions Are Addressed

Item No.	Subject/Description of Item	Section
ESP 2.5-7	Design-Related Criteria for Structural Design	2.5.4
ESP 2.5-8	Provide Ground Improvement Plans	2.5.4
ESP 2.5-9	Average Shear Wave Velocity Under Reactor Containment	2.5.4
ESP 2.5-10	Dynamic Analysis of Slope Stability	2.5.5
ESP 2.5-11	Safety Related Slopes	2.5.5
ESP 11.1-1	Offsite Doses and Maintaining Doses ALARA	11.3.1
ESP 13.6-1	Design of Protected Area Barriers	13.6
Permit Condition 3.E(1)	Exclusion Area Control	2.1.2
Permit Condition 3.E(2)	Cooling for a Second New Unit	Not applicable to Unit 3
Permit Condition 3.E(3)	Accidental Releases	2.4.13
Permit Condition 3.E(4)	Weathered or Fractured Rock	2.5.1
Permit Condition 3.E(5)	Engineered Fill	2.5.1
Permit Condition 3.E(6)	NRC Notification	2.5.1 and 2.5.4
Permit Condition 3.E(7)	Improved Soils	2.5.4

1.11 Technical Resolutions of Task Action Plan Items, New Generic Issues, New Generic Safety Issues and Chernobyl Issues

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

1.11.1 Approach

Add the following at the end of this section.

NAPS COL 1.11-1-A Table 1.11-201 supplements DCD Table 1.11-1 to address the site-specific aspects of items that refer to Notes (2) and (7).

NAPS SUP 1.11-1 Table 1.11-202 supplements DCD Table 1.11-1 to provide references to FSAR locations that provide additional information on specific issues.

1.11.2 COL Information

1.11-1-A Address Table 1.11-1 Items that refer to Notes (2) and (7)

NAPS COL 1.11-1-A This COL item is addressed in Section 1.11 and Table 1.11-201.

NAPS COL 1.11-1-A

Table 1.11-201 COL Item Resolutions Related to NUREG-0933 Table II Task Action Plan Items and New Generic Issues

Action Plan Item/ Issue Number	Description	Associated Location(s) Where Discussed and/or Technical Resolution
	Task Action Plan Items	
A-33	NEPA Review of Accident Risks	This environmental issue involves consideration of accidents on a risk specific basis. This subject is addressed in ESP-ER Chapter 7 and COLA Part 3, Chapter 7.
B-1	Environmental Technical Specifications	Issue is addressed in COLA Part 4, Sections 5.5.1 and 5.5.3, which address the Offsite Dose Calculation Manual and Radioactive Effluent Controls Program. See also Sections 11.5.4.5 and 11.5.4.6.
B-28	Radionuclide/Sediment Transport Program	Issue is addressed in COLA Part 4, Sections 5.5.1 and 5.5.3, which address the Offsite Dose Calculation Manual and Radioactive Effluent Controls Program. See also Sections 11.5.4.5 and 11.5.4.6.
B-37	Chemical Discharges to Receiving Waters	Issue is addressed in ESP-ER Section 5.3 and COLA Part 3, Sections 3.3, 3.6, and 5.2.
B-38	Reconnaissance Level Investigations	Issue is addressed in ESP-ER Chapter 2 and SSAR Chapter 2.
B-39	Transmission Lines	Issue is addressed in COLA Part 3, Sections 3.7, 4.3, and 5.6.
B-40	Effects of Power Plant Entrainment on Plankton	Issue is addressed in ESP-ER Section 5.3.1.2.
B-41	Impacts on Fisheries	Impact of power plant operation on fishery resources is addressed in ESP-ER Sections 5.3.1.2.4 and 5.3.2.2.2.
B-42	Socioeconomic Environmental Impacts	Issue is addressed in ESP-ER Sections 2.5, 4.4, and 5.8. COLA Part 3, Section 5.8 provides supplementary information on this issue.

NAPS COL 1.11-1-A

Table 1.11-201 COL Item Resolutions Related to NUREG-0933 Table II Task Action Plan Items and New Generic Issues

Action Plan Item/ Issue Number	Description	Associated Location(s) Where Discussed and/or Technical Resolution
B-43	Value of Aerial Photographs for Site Evaluation	Work completed to date on this issue is published in NUREG/CR-2861. The use of aerial photography is discussed in SSAR Sections 2.4.9, 2.5.1 and 2.5.3. Results of a visual impact study are presented in COLA Part 3, Section 5.8.
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	(3) The impact of construction and power plant operation on agricultural land use is addressed in ESP-ER Sections 4.1 and 5.1. Water use for agricultural lands is addressed in ESP-ER Sections 2.3.2 and 2.3.3. COLA Part 3 contains no additional information on this topic.
	New Generic Issues	
184	Endangered Species	Issue is addressed in ESP-ER Sections 2.4.1.6, 2.4.2.2.5, 4.3.1.2, 4.3.2, 5.3.1.2.3, 5.3.3.2, and 5.4.4. COLA Part 3 contains no additional information on this topic.

NAPS SUP 1.11-2

Table 1.11-202 Supplementary Resolutions Related to NUREG-0933 Table II TMI Action Plan Items and Human Factors Issues

Action Plan Item/ Issue Number	Description	Associated Location(s) Where Discussed and/or Technical Resolution
	TMI Action Plan Items	
1.A.1.1	Shift Technical Advisor	Sections 13.1.2.1.2.9 and DCD Section 18.6
1.A.1.2	Shift Supervisor Administrative Duties	Sections 13.1.2.1.2.5 and 13.1.2.1.2.6
1.A.1.3	Shift Manning	Section 13.1.2.1.4, Table 13.1-202, Figure 13.1-203, and DCD Section 18.6
1.A.2.1(1)	Qualifications – Experience	Section 13.1.3.1, Table 13.1-201, Section 17.5, and DCD Section 18.6
1.C.3	Shift Supervisor Responsibilities	Sections 13.1.2.1.2.5 and 13.1.2.1.2.6
1.F.2(6)	Increase the Size of Licensees' QA Staff	Table 13.1-201 and Section 17.5
1.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	Section 13.1.1.2.7, Table 13.1-201, and Section 17.5
II.B.3	Post Accident Sampling	Appendix 12BB
III.D.3.3	In-Plant Radiation Monitoring	Appendix 12BB
	Human Factors Issues	
HF1.1	Shift Staffing	Table 13.1-202 and Section 13.1.2.1.4

NAPS SUP 1.12-1

1.12 Impact of Construction Activities on Units 1 and 2

1.12.1 Introduction

Paragraph 10 CFR 52.79(a)(31) requires that the FSAR include the following information:

> For nuclear power plants to be operated on multi-unit sites, an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multi-unit sites.

Accordingly, the evaluation of the potential impact of the construction of Unit 3 on Units 1 and 2 structures, systems, and components (SSCs) important to safety is summarized below, along with a description of the managerial and administrative controls used to provide assurance that Units 1 and 2 limiting conditions for operation (LCOs) are not exceeded as a result of Unit 3 construction activities. This evaluation involves several sequential steps:

- Identification of potential construction activity hazards
- Identification of SSCs important to safety
- Identification of LCOs
- Identification of impacted SSCs and LCOs
- Identification of applicable managerial and administrative controls

1.12.2 Potential Construction Activity Hazards

Unit 3 is located on the existing NAPS site on a parcel of land adjacent to and generally west of the two operating units, Units 1 and 2, as shown in Figure 2.1-201.

Based on experience from similar projects, the scope of work necessary to construct Unit 3 is well understood. In general, it includes, but is not necessarily limited to, activities such as site exploration, grading, clearing and installation of drainage and erosion control measures; boring, drilling, dredging, demolition and excavating; storage and warehousing of equipment; and construction, erection and fabrication of new facilities. These activities involve major ESBWR standard plant structures such as the Reactor Building, Control Building, Fuel Building, Turbine Building, Radioactive Waste Building and Electrical Building; as well as related support facilities such as transformers, switchyard(s), transmission lines, cooling water structures and systems, water treatment facilities, storage tanks, etc.

The applicable time period for such activities starts when work is first performed under the COL for Unit 3 and ends for each Unit 3 SSC when responsibility for that SSC is transferred to the accountable operating organization.

Each of the types of construction activities necessary to build a new unit was examined to identify the potential hazards to the existing units. The resulting list of construction activities and potential hazards is shown in Table 1.12-201.

1.12.3 Structures, Systems and Components Important to Safety

Consistent with 10 CFR 50.34 and 10 CFR 50, Appendix A, Units 1 and 2 SSCs important to safety were identified in Chapter 3 of the NAPS Updated Final Safety Analysis Report (UFSAR) (Reference 1.12-201); additionally, information in Chapters 4, 5, 6, 7, 8 and 9 of the NAPS UFSAR was utilized.

1.12.4 Limiting Conditions for Operation

Pursuant to 10 CFR 50.36, LCOs are the lowest functional capability or performance levels of equipment required for safe operation of a facility and are established in operating unit technical specifications for each item meeting one or more of the following criteria:

- Criterion 1 – Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
- Criterion 2 – A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
- Criterion 3 – A SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
- Criterion 4 – A SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The applicable LCOs are found in the Units 1 and 2 Technical Specifications (Reference 1.12-202).

1.12.5 Impacted Structures, Systems and Components and Limiting Conditions for Operation

The information described in Sections 1.12.2–1.12.4 was evaluated to identify Units 1 and 2 SSCs and LCOs that might be impacted by Unit 3 construction activities. For example, internal/in-plant Units 1 and 2 LCO parameters such as "Shutdown Bank Insertion Limits," "RCS Minimum Temperature for Criticality" and "Secondary Specific Activity" were eliminated by examination. Similarly, SSCs both internal and specific to Units 1 and 2 are not affected. These include items such as the accumulators, fuel storage racks and rod cluster control assemblies.

For each of the potential hazards listed in Table 1.12-201, Table 1.12-202 presents the potential consequences to the SSCs of the existing units that were identified in the above process.

1.12.6 Managerial and Administrative Controls

Managerial and administrative controls are utilized to identify preventive and mitigative measures and provide notification of hazardous activity initiation in order to prevent or minimize exposure of SSCs to the identified hazards. Applicable managerial and administrative controls are listed in Table 1.12-203.

Specific hazards, impacted SSCs, and managerial and administrative controls will be developed and implemented as work progresses on site. For example, prior to construction activities that involve the use of large construction equipment such as cranes, managerial and administrative controls will be in place to prevent adverse impacts on Units 1 and 2 overhead power lines, switchyard, security boundary, etc., by providing the necessary restrictions on the use of large construction equipment.

NAPS ESP COL 2.4-1 The layout of the Unit 3 Circulating Water System (CIRC) intake and discharge piping and the construction techniques to be used for this piping will be provided to the NRC for review at least 60 days before the commencement of construction activities for this piping.

1.12.7 References

1.12-201 North Anna Power Station, Units 1 and 2, Updated Final Safety Analysis Report, Revision 38.

1.12-202 North Anna Power Station, Units 1 and 2, Technical Specifications, Amendments 231/212.

NAPS SUP 1.12-1

Table 1.12-201 Potential Hazards to Units 1 and 2 from Unit 3 Construction Activities

Construction Activity	Potential Hazards
Site Exploration, Grading, Clearing, Installation of Drainage and Erosion Control Measures, etc.	Impact on Overhead Power Lines
	Impact on Transmission Towers
	Impact on Underground Conduits, Piping, Tunnels, etc.
	Impact on Site Access and Egress
	Impact on Drainage Facilities and Structures
	Impact on Onsite Transportation Routes
	Impact on Slope Stability
	Impact of Increased Soil Erosion and Local Flooding
	Impact of Construction-Generated Dust and Equipment Exhausts
	Impact of Encroachment on Plant Protected or Vital Areas
	Impact of Encroachment on Structures and Facilities
Boring, Drilling, Pile Driving, Dredging, Demolition, Excavation, etc.	Impact on Underground Conduits, Piping, Tunnels, etc.
	Impact on Foundation Integrity
	Impact on Structural Integrity
	Impact on Slope Stability
	Impact of Ground Vibration
	Impact of Overpressure from Use of Explosives
Equipment Movement, Material Delivery, Vehicle Traffic. etc.	Impact on Overhead Power Lines
	Impact on Transmission Towers
	Impact on Underground Conduits, Piping, Tunnels, etc.
	Impact of Crane Load Drops
	Impact of Crane or Crane Boom Failures
	Impact of Vehicle Accidents
	Impact of Vehicle Runaways

NAPS SUP 1.12-1

Table 1.12-201 Potential Hazards to Units 1 and 2 from Unit 3 Construction Activities

Construction Activity	Potential Hazards
Equipment And Material Laydown, Storage, Warehousing, etc.	Impact of Releases of Stored Flammable, Hazardous or Toxic Materials
	Impact of Increase Local Flooding
	Impact of Wind-Generated, Construction-Related Debris and Missiles
General Construction, Erection, Fabrication, etc.	Impact on Instrumentation and Control Systems and Components
	Impact on Electrical Systems and Components
	Impact on Cooling Water Systems and Components
	Impact on Radioactive Waste Release Points and Parameters
	Impact of Abandonment of SSCs
	Impact of Relocation of SSCs
Connection, Integration, Tie-In, Testing, etc.	Impact on Instrumentation and Control Systems and Components
	Impact on Electrical and Power Systems and Components
	Impact on Cooling Water Systems and Components
General Site Construction Activities	Impact on Site Security Systems

NAPS SUP 1.12-1

Table 1.12-202 Potential Consequences to Units 1 and 2 Due to Potential Hazards Resulting from Unit 3 Construction Activities

Potential Hazard	Potential Consequences
Containment Structure	
Impact of Crane or Crane Boom Failures	Building Degradation Due to Crane Boom Failure
Impact of Wind-Generated Construction-Related Debris and Missiles	Effects of Construction-Related Debris or Missiles
Impact of Overpressure from Use of Explosives	Building Degradation Due to Structural Damage as a Result of Explosion
Control Room Emergency HVAC Systems	
Impact of Construction-Generated Dust and Equipment Exhausts	Effects of Construction-Generated Dust and Equipment Exhausts on Control Room Habitability Systems Air Intakes
Impact of Releases of Flammable, Hazardous or Toxic Materials	Effects of Releases of Flammable, Hazardous or Toxic Materials on Control Room Habitability Systems Design Basis
Impact of Vehicle Accidents	Effects of Releases of Flammable, Hazardous or Toxic Materials on Control Room Habitability Systems Design Basis
Diesel Generators	
Impact of Construction-Generated Dust and Equipment Exhausts	Effects of Construction-Generated Dust and Equipment Exhausts on Emergency Diesel Generator Combustion Air Intakes
Fire Protection System	
Impact on Underground Conduits, Piping, Tunnels, etc.	Degradation of FPS Availability or Capacity
Impact of the Relocation of SSCs	Degradation of FPS Availability or Capacity
Fuel Building	
Impact of Wind-Generated Construction-Related Debris and Missiles	Effects of Construction-Related Debris or Missiles
Gaseous Radioactive Waste Management System	
Impact on Radioactive Waste Release Points and Parameters	Building and Facility Effects on Gaseous Release χ/Q and D/Q Assumptions

NAPS SUP 1.12-1

Table 1.12-202 Potential Consequences to Units 1 and 2 Due to Potential Hazards Resulting from Unit 3 Construction Activities

Potential Hazard	Potential Consequences
Offsite Power System	
Impact on overhead power lines	Transmission line disruptions due to grading or clearing, equipment movement, crane boom failures, etc.
Impact on transmission towers	Transmission line disruptions due to grading or clearing, equipment movement, crane boom failures, etc.
Impact of vibratory ground motion	Operability disruptions due to vibration induced spurious trips
Impact on electrical systems and components	Operability disruptions due to equipment movement, system interconnections, etc.
Onsite Power Systems	
Impact of vibratory ground motion	Operability disruptions due to vibration induced spurious trips
Impact on electrical systems and components	Operability disruptions due to vibration induced spurious trips, system interconnections, etc.
Service Building	
Impact of crane or crane boom failures	Building degradation due to crane boom failure
Impact of wind-generated construction-related debris and missiles	Construction-related debris or missile
Service Water System	
Impact on underground conduits, piping, tunnels, etc.	Degradation of Service Water System availability or capacity
Impact on cooling water systems and structures	Degradation of Service Water System availability or capacity
Impact of the relocation of SSCs	Degradation of Service Water System availability or capacity
Ultimate Heat Sink	
Impact on underground conduits, piping, tunnels, etc.	Degradation of UHS availability or capacity
Impact on cooling water systems and components	Degradation of UHS availability or capacity

NAPS SUP 1.12-1

Table 1.12-203 Managerial and Administrative Controls for Unit 3 Construction Activity Hazards

Hazard	Control
Impact on overhead power lines	Administrative controls for appropriate standoff and/or installation of temporary support towers
Impact on transmission towers	Administrative controls for appropriate standoff and/or installation of temporary support towers
Impact on underground conduits, piping, tunnels, etc.	Administrative controls to identify potentially affected SSCs; evaluation to ensure structural integrity during construction; and/or temporary measures to mitigate impacts
Impact of construction-generated dust and equipment exhausts	Administrative controls to avoid or minimize construction dust (for example, use of water spray trucks) and/or enhanced monitoring of potentially affected system intakes, filters, etc.
Impact of overpressure from use of explosives	Administrative controls to coordinate transport, storage and use of explosives and/or temporary measures to mitigate impacts
Impact of vehicle accidents	Administrative controls to respond to site accidents (for example, construction fire brigade and/or hazardous materials response team)
Impact of ground vibration	Administrative controls to identify potentially affected SSCs, and/or temporary measures to mitigate impacts
Impact of crane or crane boom failures	Administrative controls for appropriate standoff and/or load limits (for example, minimum standoff distances and/or load limitations)
Impact of releases of flammable, hazardous or toxic materials	Administrative controls on quantities and types of flammable, hazardous or toxic materials
Impact of wind-generated, construction-related debris and missiles	Administrative controls on equipment and material storage and transport, and for reducing power or shutting down Units 1 and 2 during high winds or high wind warnings
Impact on electrical systems and components	Administrative controls to identify potentially affected SSCs; evaluation to ensure system and component integrity during construction; and/or temporary measures to mitigate impacts
Impact on cooling water systems and components	Administrative controls to identify potentially affected SSCs; evaluation to ensure system and component integrity during construction; and/or temporary measures to mitigate impacts

NAPS SUP 1.12-1

Table 1.12-203 Managerial and Administrative Controls for Unit 3 Construction Activity Hazards

Hazard	Control
Impact on radioactive waste release points and parameters	Enhanced monitoring and control to ensure releases are within limits
Impact of relocation of SSCs	Administrative controls to identify potentially affected SSCs effects of releases of flammable, hazardous or toxic materials on control room habitability systems design basis evaluation to ensure system and component integrity during construction; and/or temporary measures to mitigate impacts
Impact on site security systems	Administrative controls to coordinate construction activities with Units 1 and 2 physical protection personnel and procedures

Appendix 1A Response to TMI Related Matters

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

Table 1A-1, 10 CFR 50.34(f)(3)(i), TMI Item I.C.5

Add the following to the end of the ESBWR Resolution statement:

STD SUP 1A.1-1 ESBWR construction and operations engineers are also continually involved in reviewing industry experience from these same sources in accordance with the administrative procedures described in DCD Section 18.3.2.

Table 1A-1, 10 CFR 50.34(f)(3)(iii), TMI Item I.F.2

Add the following to the end of the ESBWR Resolution statement:

STD SUP 1A.1-1 The Quality Assurance Program described in Chapter 17 also meets the requirements of issue I.F.2 as they apply to the construction and operation of the ESBWR.

Table 1A-1, 10 CFR 50.34(f)(3)(vii), TMI Item II.J.3.1

Add "13.1" as an "Associated Location(s)" and add the following to the end of the ESBWR Resolution statement:

STD SUP 1A.1-1 The ESBWR construction and operations teams have also developed a management plan for the ESBWR project that consists of a properly structured organization with open lines of communication, clearly defined responsibilities, well-coordinated technical efforts, and appropriate control channels.

The organizational structure is discussed in Section 13.1.

Appendix 1B Plant Shielding to Provide Access to Areas and Protect Safety Equipment for Post-Accident Operation [II.B.2]

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

Appendix 1C Industry Operating Experience

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

Appendix 1C.1 Evaluation

Replace the last paragraph with the following.

STD COL 1C.1-1-A
STD COL 1C.1-2-A
STD SUP 1C-1

DCD Tables 1C-1 and 1C-2 are supplemented by Tables 1C-201 and 1C-202. These tables address Generic Letters and Bulletins that have been in effect/issued up to six months before the COL application submittal date, and after the SRP revisions that are applicable to this FSAR. They also address Generic Letter 82-39 and IE Bulletin 2005-02, which were identified in the DCD as the responsibility of the COL applicant.

Appendix 1C.2 COL Information

1C.1-1-A Handling of Safeguards Information

STD COL 1C.1-1-A

This COL item is addressed in Section 1C.1 and the Table 1C-201 entry for Generic Letter 82-39.

1C.1-2-A Emergency Preparedness and Response Actions

STD COL 1C.1-2-A

This COL item is addressed in Section 1C.1 and the Table 1C-202 entry for IE Bulletin 2005-02.

NAPS SUP 1AA.1-1

Appendix 1AA ESP Information

SSAR Chapter 1 is incorporated here by reference for historical purposes.

STD COL 1C.1-1-A

Table 1C-201 Operating Experience Review Results Summary—Generic Letters

No.	Issue Date	Title	Evaluation Result or Location(s) Where Discussed
82-39	12/22/82	Problems with the Submittals of 10 CFR 73.21 Safeguards Information Licensing Review	Not Applicable. Is an administrative communication. The site has an approved procedure for handling Safeguards Information including how to mail such information to authorized recipients.

STD COL 1C.1-2-A

Table 1C-202 Operating Experience Review Results Summary—IE Bulletins

No.	Issue Date	Title	Evaluation Result or Location(s) Where Discussed
2005-02	07/18/05	Emergency Preparedness and Response Actions for Security-Based Events	COLA Part 5, Emergency Plan

Chapter 2 Site Characteristics

2.0 Introduction

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

SSAR Sections 1.3 and 1.9 are incorporated by reference for historical purposes only.

Replace the last two paragraphs with the following paragraphs.

NAPS COL 2.0-1-A

DCD site parameter values for the ESBWR standard plant are identified in DCD Table 2.0-1 and DCD Tier 1, Table 5.1-1.

ESP site characteristic values are identified in Appendix A of the ESP (Reference 2.0-203). The ESP design parameter values are identified as controlling values of parameters and design basis accident source term plant parameters in Appendix B of the ESP.

Table 2.0-201 provides several evaluations:

- Part 1 of Table 2.0-201 identifies each DCD site parameter value and the corresponding ESP and Unit 3 site characteristic values. In accordance with 10 CFR 52.79(b) and (d); and SRP Section 2.0, Part 1 of Table 2.0-201 evaluates, as applicable, whether:
  - ESP site characteristic values fall within DCD site parameter values
  - Unit 3 site characteristic values fall within DCD site parameter values
  - Unit 3 site characteristic values fall within ESP site characteristic values

NAPS SUP 2.0-1

- Part 2 of Table 2.0-201 identifies those ESP site characteristics and design parameters for which there is no corresponding DCD site parameter value. In accordance with 10 CFR 52.79(b) and SRP Section 2.0, Part 2 of Table 2.0-201 evaluates whether the Unit 3 site characteristic or facility design value falls within the ESP site characteristic or ESP design parameter value.

NAPS SUP 2.0-2

- Part 3 of Table 2.0-201 identifies those site characteristics and design parameters listed in SSAR Table 1.9-1 for which there is not already a comparison to a corresponding DCD or ESP value in the first two parts of Table 2.0-201. In accordance with the commitment in

SSAR Section 1.3, Part 3 of Table 2.0-201 evaluates whether the Unit 3 site characteristic or facility design value falls within the SSAR Table 1.9-1 site characteristic or design parameter value. (Some site characteristic and design parameter values listed in SSAR Table 1.9-1 are included in the evaluation in Parts 1 and 2 of Table 2.0-201.)

Appendix 2A provides site-specific input values used in ARCON96 analyses of on-site χ/Q values.

NAPS COL 2.0-2-A through 2.0-30-A

Information on Unit 3 site characteristics is provided in Sections 2.1 through 2.5, which incorporate by reference, the corresponding SSAR sections. This information addresses NRC guidance in NUREG-0800 as identified in Table 2.0-2R. In the "COL Information" column, the COL Item from the DCD is replaced with information responding to the COL Item and identifying the FSAR section which addresses the SRP section invoked by the COL Item.

2.0.1 COL Information

2.0-1-A Site Characteristics Demonstration

NAPS COL 2.0-1-A

This COL item is addressed in Section 2.0.

2.0-2-A through 2.0-30-A Standard Review Plan Conformance

NAPS COL 2.0-2-A through 2.0-30-A

These COL items are addressed in Section 2.0.

2.0.2 References

2.0-201 [Deleted]

2.0-202 NUREG-1835, Safety Evaluation Report for an Early Site Permit (ESP) at the North Anna ESP Site, U.S. Nuclear Regulatory Commission, September 2005.

2.0-203 Early Site Permit (ESP) for the North Anna ESP Site, No. ESP-003, U.S. Nuclear Regulatory Commission, November 2007.

NAPS COL 2.0-2-A through 2.0-30-A

Table 2.0-2R Limits Imposed on Acceptance Criteria in Section II of SRP by ESBWR Design

	Section	Subject	ESBWR DCD Parameters, Considerations and/or Limits	COL Information
NAPS COL 2.0-2-A	2.1.1	Site Location and Description	None	COL Item 2.0-2-A is addressed in Section 2.1.1.
NAPS COL 2.0-3-A	2.1.2	Exclusion Area Authority and Control	None	COL Item 2.0-3-A is addressed in Section 2.1.2.
NAPS COL 2.0-4-A	2.1.3	Population Distribution	ESBWR PRA offsite consequence analysis in DCD Reference 2.0-1 is based on a population density of 305 people per square kilometer (790 per square mile).	COL Item 2.0-4-A is addressed in Section 2.1.3. The population density for offsite analysis provided in Section 2.1.3 falls within (is less than) the density used in DCD Reference 2.0-1.
NAPS COL 2.0-5-A	2.2.1–2.2.2	Identification of Potential Hazards in Site Vicinity	Per DCD Table 2.0-1	COL Item 2.0-5-A is addressed in Section 2.2.
NAPS COL 2.0-6-A	2.2.3	Evaluation of Potential Accidents	None considered in vicinity of plant	COL Item 2.0-6-A is addressed in Section 2.2.3.
NAPS COL 2.0-7-A	2.3.1	Regional Climatology	Per DCD Table 2.0-1	The portion of COL Item 2.0-7-A to provide information in accordance with SRP 2.3.1 is addressed in Section 2.3.1. The wind speed used in design of nonsafety-related structures that are not included as part of the ESBWR Standard Plant design is 40 m/s (90 mph).
NAPS COL 2.0-8-A	2.3.2	Local Meteorology	None	COL Item 2.0-8-A is addressed in Section 2.3.2.
NAPS COL 2.0-9-A	2.3.3	Onsite Meteorological Measurements Programs	None	COL Item 2.0-9-A is addressed in Section 2.3.3.

NAPS COL 2.0-2-A through 2.0-30-A

Table 2.0-2R Limits Imposed on Acceptance Criteria in Section II of SRP by ESBWR Design

	Section	Subject	ESBWR DCD Parameters, Considerations and/or Limits	COL Information
NAPS COL 2.0-10-A	2.3.4	Short-Term Dispersion Estimates for Accidental Atmospheric Releases	Per DCD Table 2.0-1. See also Chapter 15.	The portion of COL Item 2.0-10-A to supply information in accordance with SRP 2.3.4 is addressed in Section 2.3.4. Information provided in Table 2.0-201 shows that the site characteristic short-term meteorological dispersion values fall within the site parameter values. This means that dose values given in DCD Chapter 15 remain bounding for this FSAR and less than stipulated in 10 CFR 50.34(a) and the applicable portions of SRP Sections 11 and 15.
NAPS COL 2.0-11-A	2.3.5	Long-Term Diffusion Estimates	Per DCD Table 2.0-1. See Sections 2.3.5 and 12.2.2.1 for a discussion of the generation of these values.	COL Item 2.0-11-A is addressed in Section 2.3.5.
NAPS COL 2.0-12-A	2.4.1	Hydraulic Description Maximum Groundwater Level	Per DCD Table 2.0-1	COL Item 2.0-12-A is addressed in Section 2.4.1.
NAPS COL 2.0-13-A	2.4.2	Floods	Per DCD Table 2.0-1	COL Item 2.0-13-A is addressed in Section 2.4.2.
NAPS COL 2.0-14-A	2.4.3	Probable Maximum Flood on Streams and Rivers	Probable maximum flooding level on streams and rivers does not exceed the maximum flood level defined in DCD Table 2.0-1.	COL Item 2.0-14-A is addressed in Section 2.4.3.
NAPS COL 2.0-15-A	2.4.4	Potential Dam Failures	Potential dam failures do not cause flooding to exceed the maximum flood level defined in DCD Table 2.0-1.	COL Item 2.0-15-A is addressed in Section 2.4.4.

NAPS COL 2.0-2-A through 2.0-30-A

Table 2.0-2R Limits Imposed on Acceptance Criteria in Section II of SRP by ESBWR Design

	Section	Subject	ESBWR DCD Parameters, Considerations and/or Limits	COL Information
NAPS COL 2.0-16-A	2.4.5	Probable Maximum Surge and Seiche Flooding	Probable maximum surge and seiche flooding level does not exceed the maximum flood level defined in DCD Table 2.0-1.	COL Item 2.0-16-A is addressed in Section 2.4.5.
NAPS COL 2.0-17-A	2.4.6	Probable Maximum Tsunami Flooding	Probable maximum tsunami flooding level does not exceed the maximum flood level defined in DCD Table 2.0-1.	COL Item 2.0-17-A is addressed in Section 2.4.6.
NAPS COL 2.0-18-A	2.4.7	Ice Effects	None	COL Item 2.0-18-A is addressed in Section 2.4.7.
NAPS COL 2.0-19-A	2.4.8	Cooling Water Canals and Reservoirs	None	COL Item 2.0-19-A is addressed in Section 2.4.8.
NAPS COL 2.0-20-A	2.4.9	Channel Diversions	None	COL Item 2.0-20-A is addressed in Section 2.4.9.
NAPS COL 2.0-21-A	2.4.10	Flooding Protection Requirements	None	COL Item 2.0-21-A is addressed in Section 2.4.10.
NAPS COL 2.0-22-A	2.4.11	Cooling Water Supply	None	COL Item 2.0-22-A is addressed in Section 2.4.11.
NAPS COL 2.0-23-A	2.4.12	Groundwater	Per DCD Table 2.0-1	COL Item 2.0-23-A is addressed in Section 2.4.12.
NAPS COL 2.0-24-A	2.4.13	Accidental Releases of Liquid Effluents in Ground and Surface Waters	The source term provided in DCD Table 12.2-13a, “Liquid Waste Management System Equipment Drain Collection Tank Activity,” is used in the effects analysis.	COL Item 2.0-24-A is addressed in Section 2.4.13.
NAPS COL 2.0-25-A	2.4.14	Technical Specifications and Emergency Operation Requirements	None	COL Item 2.0-25-A is addressed in Section 2.4.14.

NAPS COL 2.0-2-A through 2.0-30-A

Table 2.0-2R Limits Imposed on Acceptance Criteria in Section II of SRP by ESBWR Design

	Section	Subject	ESBWR DCD Parameters, Considerations and/or Limits	COL Information
NAPS COL 2.0-26-A	2.5.1	Basic Geologic and Seismic Information	None	COL Item 2.0-26-A is addressed in Section 2.5.1.
NAPS COL 2.0-27-A	2.5.2	Vibratory Ground Motion	Per DCD Table 2.0-1 (and DCD Figures 2.0-1 and 2.0-2)	The portion of COL Item 2.0-27-A to provide information in accordance with SRP 2.5.2 is addressed in Section 2.5.2. Information provided in Table 2.0-201 confirms that reactor building/fuel building (RB/FB), control building (CB), and firewater service complex (FWSC) foundation input response spectra (FIRS) are enveloped by the ESBWR certified seismic design response spectra (CSDRS) referenced at foundation level.
NAPS COL 2.0-28-A	2.5.3	Surface Faulting	ESBWR design assumes no permanent ground deformation from tectonic or non-tectonic faulting.	COL Item 2.0-28-A is addressed in Section 2.5.3. Information to address permanent ground deformation from tectonic or non-tectonic faulting is provided in Section 2.5.3.
NAPS COL 2.0-29-A	2.5.4	Stability of Subsurface Materials and Foundations	Per DCD Table 2.0-1	The portion of COL Item 2.0-29-A to provide information in accordance with SRP 2.5.4 is addressed in Section 2.5.4. Information to address localized liquefaction potential under other than Seismic Category I structures is provided in Section 2.5.4.8. Information to address settlements and differential settlements is provided in Section 2.5.4.10.2.
NAPS COL 2.0-30-A	2.5.5	Stability of Slopes	Per DCD Table 2.0-1	COL Item 2.0-30-A is addressed in Section 2.5.5.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Maximum Groundwater Level	0.61 m (2 ft) below plant grade		The DCD site parameter of maximum groundwater level of 0.61 m (2 ft) below plant grade is the same as the design groundwater level in DCD Table 3.4-1. The design plant grade elevation identified in DCD Table 3.4-1 is at 4650 mm, which corresponds to 88.4 m (290 ft) msl for the Unit 3 site as shown in Figure 2.1-201. Therefore, the DCD site parameter value of 0.61 m (2 ft) below plant grade corresponds to a maximum groundwater level no higher than 87.8 m (288 ft) msl for the Unit 3 site.
			ESP 82.3 m (270 ft) msl or 0.3 m (1 ft) below the free surface, whichever is higher	The ESP site characteristic value for maximum groundwater level is defined in ESP, Appendix A, as the maximum elevation of groundwater at the ESP site. The ESP value of 82.3 m (270 ft) msl is based on the proposed site grade in the SSAR of 82.6 m (271 ft) msl. With design plant grade for Unit 3 at 88.4 m (290 ft) msl, the operative ESP site characteristic value becomes 0.3 m (1 ft) below the free surface which is higher than 82.3 m (270 ft) msl. With a free surface at 88.4 m (290 ft) msl, the ESP site characteristic corresponds to 88.1 m (289 ft) msl which does not fall within (is higher than) the value established by the DCD site parameter. SSAR Table 1.9-1 provides a value of < 82.3 m (270 ft) msl from SSAR Section 2.4.12.4 which is based on the proposed site grade in the SSAR of 82.6 m (271 ft) msl.
			Unit 3 2.1 m (7 ft) below design plant grade	The Unit 3 site characteristic value for maximum groundwater level below design plant grade is 2.1 m (7 ft) in the power block area based on the maximum groundwater elevation of 86.3 m (283 ft) msl from Section 2.4.12 and the design plant grade elevation of 88.4 m (290 ft) msl. Therefore, the Unit 3 site characteristic value for maximum groundwater level below design plant grade falls within (is lower than) the DCD site parameter value. The maximum groundwater level in the power block area is 2.1 m (7 ft) below design plant grade, which meets the DCD site parameter limit of not higher than 0.61 m (2 ft) below design plant grade. The Unit 3 site characteristic value falls within (is lower than) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Extreme Wind			
	Seismic Category I and II Structures			
	100-year Wind Speed (3-sec gust) [13]	67.1 m/s (150 mph)	**ESP and Unit 3** 42.9 m/s (96 mph), 3-second gust	The ESP site characteristic value for basic wind speed is defined as the 3-second gust wind speed at 10 m (33 ft) above the ground that has a 1 percent annual probability of being exceeded (100-year mean recurrence interval). The ESP site characteristic value for basic wind speed falls within (is lower than) the DCD site parameter value. SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.1, provides the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.
	Exposure Category	D		The DCD site parameter of extreme wind exposure category is determined using ASCE 7 (DCD Reference 2.0-2). Exposure category is determined by a number of variables including wind speed, building shape and location, and surface roughness. A DCD site parameter of Exposure Category D results in the most severe design wind pressures.
			ESP No value provided	
			Unit 3 Exposure Category D	The Unit 3 site characteristic is Exposure Category D as this value cannot be exceeded. The Unit 3 site characteristic falls within (is the same as) the DCD site parameter value for extreme wind exposure category, i.e., Exposure Category D.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Extreme Wind (continued)			
	Non-Seismic Standard Plant Structures			
	50-year Wind Speed (3-sec gust)	58.1 m/s (130 mph)	**ESP** No value provided	
			Unit 3 42.9 m/s (96 mph) wind speed, 3-second gust, with a 100-year recurrence interval	The Unit 3 site characteristic value is the same as the ESP and Unit 3 site characteristic value for a 100-year wind speed (3-sec gust) identified above. This ESP and Unit 3 value is 42.9 m/s (96 mph). This value falls within (is less than) the DCD site parameter value for the 50-year wind speed (3-sec gust) of 58.1 m/s (130 mph). Because the 50-year wind speed (3-sec gust) value at Unit 3 can not be higher than the 100-year wind speed (3-sec gust), the Unit 3 site characteristic value for 50-year wind speed (3-sec gust) also falls within (is lower than) the DCD site parameter value for 50-year wind speed (3-sec gust). SSAR Section 2.3.1.3.1 provides the same value for a 100-year wind speed (3-sec gust) as ESP, Appendix A.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Maximum Flood (or Tsunami) Level [2]	0.3 m (1 ft) below plant grade		The DCD site parameter of maximum flood (or tsunami) water level of 0.3 m (1 ft) below plant grade is the same as the design flood level in DCD Table 3.4-1. The design plant grade elevation identified in DCD Table 3.4-1 is at 4650 mm, which corresponds to 88.4 m (290 ft) msl for the Unit 3 site as shown in Figure 2.1-201. Therefore, the DCD site parameter value of 0.3 m (1 ft) below plant grade corresponds to a maximum flood water level below 88.1 m (289 ft) msl for the Unit 3 site.
			ESP 82.3 m (270 ft) msl based on PMF	The ESP site characteristic value for maximum flood water level is defined as the maximum flood level at the ESP site due to a probable maximum flood (PMF) in Lake Anna's watershed, simultaneous failure of upstream storage reservoirs, and coincident wind-wave action. This value is 82.3 m (270 ft) msl at the Unit 3 site based on the PMF and remains the same value after the increase in design plant grade for Unit 3 to 88.4 m (290 ft) msl. The ESP site characteristic value falls within (is lower than) the DCD site parameter value.
			Unit 3 0.85 m (2.8 ft) below design plant grade based on PMP	The Unit 3 site characteristic value for PMF of 81.5 m (267.39 ft) msl is provided in SSAR Section 2.4.3 and SSAR Table 1.9-1, and falls within (is less than) the DCD site parameter value and the ESP site characteristic value. The Unit 3 site characteristic value for maximum flood water level below design plant grade is due to the local probable maximum precipitation (PMP) flood. As described in Section 2.4.2, this value is 0.85 m (2.8 ft) below design plant grade in the power block area based on the local PMP flood water elevation of 87.54 m (287.2 ft) msl in this area. Therefore, the Unit 3 site characteristic value for maximum flood water level below design plant grade falls within (is lower than) the DCD site parameter value. The maximum flood water level in the power block area due to local PMP is 0.85 m (2.8 ft) below design plant grade, which meets the DCD site parameter limit for a maximum flood water level not higher than 0.3 m (1 ft) below design plant grade.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Tornado			
	Maximum Tornado Wind Speed (3)	147.5 m/s (330 mph)	**ESP and Unit 3** 116.2 m/s (260 mph)	The ESP site characteristic value for design basis tornado maximum wind speed is defined as the maximum wind speed resulting from passage of a tornado having a probability of occurrence of 10^{-7} per year. The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.2, provides the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.
	Maximum Rotational Speed	116.2 m/s (260 mph)	**ESP and Unit 3** 93.0 m/s (208 mph)	The ESP site characteristic value for design basis tornado maximum rotational speed is defined as the rotation component of the maximum tornado wind speed. The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.2, provides the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.
	Translational Speed	31.3 m/s (70 mph)	**ESP and Unit 3** 23.2 m/s (52 mph)	The ESP site characteristic value for design basis tornado maximum translational speed is defined as the translational component of the maximum tornado wind speed. The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.2, provides the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1] [17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Tornado (continued)			
	Radius	45.7 m (150 ft)	**ESP and Unit 3** 45.7 m (150 ft)	The ESP site characteristic value for design basis tornado radius of maximum rotational speed is defined as the distance from the center of the tornado at which the maximum rotational wind speed occurs. The ESP site characteristic value falls within (is the same as) the DCD site parameter value. SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.2, provides the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.
	Pressure Drop	16.6 kPa (2.4 psi)	**ESP and Unit 3** 10.3 kPa (1.5 psi)	The ESP site characteristic value for design basis tornado pressure drop is defined as the decrease in ambient pressure from normal atmospheric pressure resulting from passage of the tornado. The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.2, provides the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.
	Rate of Pressure Drop	11.7 kPa/s (1.7 psi/s)	**ESP and Unit 3** 5.2 kPa/s (0.76 psi/s)	The ESP site characteristic value for design basis tornado maximum rate of pressure drop is defined as the rate of pressure drop resulting from the passage of the tornado. The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.2, provides the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Tornado (continued)			
Missile Spectrum (3)	Spectrum I of SRP 3.5.1.4, Rev. 2 applied to full building height.	**ESP** No value provided	The DCD site parameter for tornado missile spectrum is based on SRP 3.5.1.4, Rev. 2, July 1981, with Spectrum I missiles applied to full building height. When the missiles in Spectrum I are applied to full building height and not limited to impacts at altitudes less than 9.1 m (30 ft) above all grade levels within 0.8 km (0.5 mi) of the safety-related structures, the DCD site parameter addresses variations in grade levels at a site.
		Unit 3 Spectrum I of SRP 3.5.1.4, Rev. 2 applied to full building height	The Unit 3 site characteristic for tornado missile spectrum is Spectrum I of SRP 3.5.1.4, Rev. 2, applied to full building height. This spectrum fully addresses variations in grade levels at the Unit 3 site and this Unit 3 site characteristic value falls within (is the same as) the DCD site parameter value for tornado missile spectrum.
Precipitation (for Roof Design)			
Maximum Rainfall Rate (4)	49.3 cm/hr (19.4 in/hr)	**ESP** 46.5 cm (18.3 in)/hr	The ESP site characteristic value for local intense precipitation is defined as the maximum potential rainfall at the immediate ESP site in inches of rain in an hour. This value is 46.5 cm (18.3 in)/hr. The ESP site characteristic value falls within (is less than) the DCD site parameter value.
		Unit 3 46.5 cm/hr (18.3 in/hr)	The Unit 3 site characteristic value of 46.5 cm/hr (18.3 in/hr) is from SSAR Table 2.4-3 and SSAR Table 1.9-1, and falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Precipitation (for Roof Design) (continued)			
	Maximum Short Term Rate	15.7 cm (6.2 in) in 5 min	**ESP** 15.5 cm (6.1 in) in 5 min	The ESP site characteristic value for local intense precipitation is defined as the maximum potential rainfall at the immediate ESP site in inches of rain in five minutes. This value is 15.5 cm (6.1 in) inches in 5 minutes. The ESP site characteristic value falls within (is less than) the DCD site parameter value.
			Unit 3 15.5 cm (6.1 in) in 5 min	The Unit 3 site characteristic value of 15.5 cm (6.1 in) in 5 min is from SSAR Table 2.4-3 and SSAR Table 1.9-1, and falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Precipitation (for Roof Design) (continued)			
	Maximum Roof Load [5]	2873 Pa (60 lbf/ft^2)	**ESP** No value provided	
			Unit 3 2121 Pa (44.3 lbf/ft^2)	The Unit 3 site characteristic value for maximum roof load is based on site characteristic values for both 100-yr snow pack and 48-hr PMWP, each of which are less than the corresponding DCD site parameter value (as shown in comparisons below). The Unit 3-specific roof live load from antecedent snow pack represents a 100-year return ground snow load of 1460 Pa (30.5 lb/sq ft) that on the roof of each safety-related building is taken as 60% of that value based on exposure and thermal conditions per the ASCE 7 Commentary in DCD Reference 2.0-2. Therefore, the roof snow load from the antecedent snow pack is no more than 876 Pa (18.3 lbf/ft^2) for any Unit 3 safety-related building. Also, as described in DCD Table 3G.1-2, the roof scuppers and drains are designed independently to handle the 48-hr probable maximum winter precipitation (PMWP) with no more than 100 mm (4 in) of water accumulation on the roof. The added load from such an accumulation is no more than 1005 Pa (21 lbf/ft^2) for any safety-related Unit 3 building. Because precipitation during a PMWP event is liquid at the North Anna site, the total roof loading includes a rain-on-snow surcharge to account for liquid flowing through the 100-yr snow pack on the roof before it accumulates on the roof. Per Section 7.10 of ASCE 7, 239 Pa (5 lbf/ft^2) accounts for the rain-on-snow surcharge. Therefore, the total maximum roof load (snow pack plus rain) on a Unit 3 safety-related building is 2121 Pa (18.3 + 21 + 5 or 44.3 lbf/ft^2). The Unit 3 site characteristic value of 2121 Pa (44.3 lbf/ft^2) falls within (is lower than) the DCD site parameter value of 2873 Pa (60 lbf/ft^2).

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Precipitation (for Roof Design) (continued)			
Maximum Ground Snow Load(5) (100-year recurrence interval):	2394 Pa (50 lbf/ft^2)	**ESP and Unit 3** 1460 Pa (30.5 lb/ft^2) (100-yr recurrence)	The ESP site characteristic value for maximum ground snow load is defined as the weight of the 100-yr return period snow pack (to be used in determining extreme winter precipitation loads for roofs). The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Section 2.3.1.3.4 and SSAR Table 1.9-1 provide the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.
Maximum 48-hr Winter Rainfall (5)	91.4 cm (36 in)	**ESP and Unit 3** 52.7 cm (20.75 in) of water (48-hr probable maximum winter precipitation)	The ESP site characteristic value for 48-hr probable maximum winter precipitation is defined as the probable maximum precipitation during the winter months (to be used in conjunction with the 100-year snow pack in determining extreme winter precipitation loads for roofs). The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Section 2.3.1.3.4 and SSAR Table 1.9-1 provide the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Ambient Design Temperature[6]			
	2% Annual Exceedance Values			
	Maximum	35.6ºC (96ºF) dry bulb 26.1ºC (79ºF) wet bulb (mean coincident)	**ESP and Unit 3** 32.2ºC (90ºF) dry bulb with 23.9ºC (75ºF) wet bulb (mean coincident) (2% annual exceedance values)	The ESP site characteristic values for maximum dry-bulb temperature with mean coincident wet-bulb temperature for 2% annual exceedance are the ambient dry-bulb temperature (and mean coincident wet-bulb temperature) that will be exceeded 2% of the time annually. The ESP site characteristic values fall within (are lower than) the DCD site parameter values. SSAR Table 2.3-18 and SSAR Table 1.9-1 provide the same values as ESP, Appendix A. The Unit 3 site characteristic values fall within (are the same as) the ESP site characteristic values.
		27.2ºC (81ºF) wet bulb (non-coincident)	**ESP** No value provided	
			Unit 3 26.1ºC (79ºF) wet bulb (non-coincident) (0.4% annual exceedance value)	The Unit 3 site characteristic value is the ESP site characteristic value for the maximum wet bulb temperature (non-coincident) for 0.4% annual exceedance. This value is defined as the ambient wet-bulb temperature that will be exceeded 0.4% of the time annually. This value is 26.1ºC (79ºF) wet bulb (non-coincident) and falls within (is less than) the DCD site parameter value for 2% annual exceedance. Because the 2% site characteristic value is even lower than the 0.4% value, the site's 2% value also falls within (is lower than) the DCD site parameter value for 2% annual exceedance. SSAR Table 2.3-18 and SSAR Table 1.9-1 provide the same 0.4% value as ESP, Appendix A.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Ambient Design Temperature (continued) 2% Annual Exceedance Values (continued)			
	Minimum	–23.3ºC (–10ºF)	**ESP** No value provided	
			Unit 3 –7.8ºC (18ºF) (99% annual exceedance value)	The Unit 3 site characteristic value is the ESP site characteristic value for the minimum dry bulb temperature for 99% annual exceedance. This value is defined as the ambient dry-bulb temperature below which dry-bulb temperatures will fall 1% of the time annually. This value is –7.8ºC (18ºF) and falls within (is higher than) the DCD site parameter value for 2% annual exceedance (i.e., the ambient dry-bulb temperature below which dry-bulb temperatures will fall 2% of the time annually). Because the minimum temperature site characteristic value for 2% is even higher than the 1% value, the site's 2% value also falls within (is higher than) the DCD site parameter value for 2% annual exceedance. SSAR Table 2.3-18 and SSAR Table 1.9-1 provide the same 1% value as ESP, Appendix A.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Ambient Design Temperature (continued)			
1% Annual Exceedance Values			
Maximum	37.8ºC (100ºF) dry bulb 26.1ºC (79ºF) wet bulb (mean coincident)	**ESP** No value provided	
		Unit 3 35ºC (95ºF) dry bulb with 25ºC (77ºF) wet bulb (mean coincident) (0.4% annual exceedance value)	The Unit 3 site characteristic values are the ESP site characteristic values for the maximum dry bulb temperature with mean coincident wet bulb temperature for 0.4% annual exceedance. These values are the ambient dry-bulb temperature (and mean coincident wet-bulb temperature) that will be exceeded 0.4 percent of the time annually. These values are 35°C (95°F) dry bulb with 25°C (77°F) wet bulb (mean coincident) and fall within (are less than) the DCD site parameter values for 1% exceedance. Because the 1% site characteristic values are even lower than the 0.4% values, the site's 1% values also fall within (are lower than) the DCD site parameter values for 1% annual exceedance. SSAR Table 2.3-18 and SSAR Table 1.9-1 provide the same 0.4% values as ESP, Appendix A.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Ambient Design Temperature (continued) **1% Annual Exceedance Values (continued)**			
Maximum	27.8°C (82°F) wet bulb (non-coincident)	**ESP** No value provided	
		Unit 3 26.1°C (79°F) wet-bulb (non-coincident) (0.4% annual exceedance value)	The Unit 3 site characteristic value is the ESP site characteristic value for the maximum wet bulb temperature (non-coincident) for 0.4% annual exceedance. This value is defined as the ambient wet-bulb temperature that will be exceeded 0.4% of the time annually. This value is 26.1°C (79°F) wet bulb (non-coincident) and falls within (is less than) the DCD site parameter value for 1% annual exceedance. Because the 1% site characteristic value is even lower than the 0.4% value, the site's 1% value also falls within (is lower than) the DCD site parameter value for 1% annual exceedance. SSAR Table 2.3-18 and SSAR Table 1.9-1 provide the same 0.4% value as ESP, Appendix A.
Minimum	-23.3°C (-10°F)	**ESP and Unit 3** –7.8°C (18°F) (99% annual exceedance value)	The ESP site characteristic value for minimum dry-bulb temperature 99% annual exceedance is defined as the ambient dry-bulb temperature below which dry-bulb temperatures will fall 1% of the time annually. The ESP site characteristic value falls within (is higher than) the DCD site parameter value. SSAR Table 2.3-18 and SSAR Table 1.9-1 provide the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Ambient Design Temperature (continued)			
	0% Exceedance Values			
	Maximum	47.2ºC (117ºF) dry bulb 26.7ºC (80ºF) wet bulb (mean coincident)	**ESP** No value provided	
			Unit 3 42.8°C (109°F) dry-bulb with 24.4°C (76°F) wet bulb coincident (100-year return values)	The Unit 3 site characteristic values for maximum dry bulb with coincident wet bulb temperatures are the maximum dry bulb temperature for a 100-year return period as provided in SSAR Tables 2.3-18 and 1.9-1, and its corresponding wet bulb temperature (using a correlation between dry bulb and wet bulb temperatures). As shown in Section 2.3.1.2, these values are 42.8°C (109°F) dry-bulb with 24.4°C (76°F) wet bulb coincident and fall within (are less than) the DCD site parameter values for 0% exceedance. The Unit 3 site characteristic 0% exceedance values (historic maximum values) for dry bulb with coincident wet bulb temperatures are provided in SSAR Tables 2.3-18 and 1.9-1, and also fall within (are less than) the DCD site parameter values for 0% exceedance.
		31.1ºC (88ºF) wet bulb (non-coincident)	**ESP** No value provided.	
			Unit 3 31.1°C (88°F) wet-bulb (non-coincident) (100-year return value)	The Unit 3 site characteristic value for maximum wet bulb temperature (non-coincident) is the 100-year return period temperature as provided in SSAR Tables 2.3-18 and 1.9-1. This value is 31.1°C (88°F) wet bulb non-coincident and falls within (is equal to) the DCD site parameter value for 0% exceedance. The Unit 3 site characteristic 0% exceedance value (historic maximum value) for wet bulb temperature (non-coincident) is provided in SSAR Tables 2.3-18 and 1.9-1, and also falls within (is less than) the DCD site parameter value for 0% exceedance.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Ambient Design Temperature (continued) **0% Exceedance Values (continued)**			
	Minimum	–40°C (–40°F)	**ESP** No value provided	
			Unit 3 –29.4°C (–21°F) (0% exceedance value)	The Unit 3 site characteristic value for minimum 0% exceedance value temperature is the historic minimum dry bulb temperature as provided in SSAR Table 2.3-5. This value is –29.4°C (–21°F) and falls within (is higher than) the DCD site parameter value for 0% exceedance.
	Soil Properties[16] **Minimum Static Bearing Capacity**[7]			
	Reactor/Fuel Building	699 kPa (14,600 lbf/ft^2)		The DCD site parameter of minimum static bearing capacity underlying the reactor building/fuel building foundation is determined by the minimum static bearing capacity for any layer of material under this foundation. As shown in Table 2.5-215, concrete fill, Zone III-IV, and Zone IV materials are under the reactor building/fuel building foundation for Unit 3. Of these, the Zone III-IV material has the lowest minimum bearing capacity value.
			ESP and Unit 3 3830 kPa (80,000 lbf/ft^2) for Zone III-IV material	The ESP site characteristic value for minimum bearing capacity of Zone III-IV material is defined as the allowable load-bearing capacity of this layer for supporting plant structures. This value is 3830 kPa (80,000 lbf/ft^2) and falls within (is greater than) the DCD site parameter value. SSAR Section 2.5.4 provides the same value as ESP, Appendix A. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Soil Properties[16] (continued) **Minimum Static Bearing Capacity (continued)**			
	Control Building	292 kPa (6,100 lbf/ft^2)		The DCD site parameter of minimum static bearing capacity underlying the control building foundation is determined by the minimum static bearing capacity for any layer of material under this foundation. As shown in Table 2.5-215, concrete fill, Zone III, Zone III-IV, and Zone IV materials are under the control building foundation for Unit 3. Of these, the Zone III material has the lowest minimum bearing capacity value.
			ESP 766 kPa (16,000 lbf/ft^2) for Zone III weathered rock	The ESP site characteristic value for minimum bearing capacity of Zone III material is defined as the allowable load-bearing capacity of this layer for supporting plant structures. This value is 766 kPa (16,000 lbf/ft^2) and falls within (is greater than) the DCD site parameter value.
			Unit 3 2394 kPa (50,000 lbf/ft^2) for the mean of Zone III and Zone III-IV materials	The Unit 3 site characteristic value of minimum static bearing capacity for materials underlying the control building is from Section 2.5.4.10.1.c and is the mean of the values for Zone III and Zone III-IV materials beneath the control building. The Unit 3 site characteristic value for minimum bearing capacity of Zone III material is 958 kPa (20,000 lbf/ft^2). The Unit 3 site characteristic value for minimum bearing capacity of Zone III-IV material is 3830 kPa (80,000 lbf/ft^2). The mean of the values for Zone III and Zone III-IV materials beneath the control building is 2394 kPa (50,000 lbf/ft^2). The Unit 3 site characteristic value for Zone III and the mean of the values for Zone III and Zone III-IV materials each fall within (is greater than) the DCD site parameter value. The Unit 3 site characteristic value for Zone III falls within (is greater than) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Soil Properties[16] (continued) **Minimum Static Bearing Capacity (continued)**			
	Firewater Service Complex	165 kPa (3450 lbf/ft^2)		The DCD site parameter of minimum static bearing capacity underlying the FWSC foundation is determined by the minimum static bearing capacity for any layer of material under this foundation. As shown in Table 2.5-215, structural fill, Zone III, Zone III-IV, and Zone IV materials are under the FWSC foundation for Unit 3. Of these, the Zone III material has the lowest minimum bearing capacity value.
			ESP 766 kPa (16,000 lbf/ft^2) for Zone III weathered rock	The ESP site characteristic value for minimum bearing capacity of Zone III material is defined as the allowable load-bearing capacity of this layer for supporting plant structures. This value is 766 kPa (16,000 lbf/ft^2) and falls within (is greater than) the DCD site parameter value.
			Unit 3 958 kPa (20,000 lbf/ft^2) for Zone III weathered rock	The Unit 3 site characteristic value for minimum bearing capacity of Zone III material is 958 kPa (20,000 lbf/ft^2). The Unit 3 site characteristic value for Zone III falls within (is greater than) the DCD site parameter value. The Unit 3 site characteristic value for Zone III falls within (is greater than) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Soil Properties[16] (continued) **Minimum Dynamic Bearing Capacity (continued)**			
	Reactor/Fuel Building			
	Soft	2700 kPa (56,400 lbf/ft^2)	**ESP** No values provided	
	Medium	7300 kPa (152,500 lbf/ft^2)	**Unit 3** 10,250 kPa (214,000 lbf/ft^2)	The Unit 3 site characteristic value for minimum dynamic bearing capacity for the RB/FB structure is from Table 2.5-215 and falls within (is greater than) the DCD site parameter minimum value for any type of soil: hard, medium, or soft. Based on the equivalent uniform shear wave velocity identified below, the materials beneath the RB/FB structure are classified as hard in accordance with Note (7).
	Hard	5400 kPa (112,800 lbf/ft^2)		
	Control Building			
	Soft	2800 kPa (58,500 lbf/ft^2)	**ESP** No values provided	
	Medium	2500 kPa (52,300 lbf/ft^2)	**Unit 3** 6895 kPa (144,000 lbf/ft^2)	The Unit 3 site characteristic value for minimum dynamic bearing capacity for the CB structure is from Table 2.5-215 and falls within (is greater than) the DCD site parameter minimum value for any type of soil: hard, medium, or soft. Based on the equivalent uniform shear wave velocity identified below, the materials beneath the CB structure are classified as hard in accordance with Note (7).
	Hard	2400 kPa (50,200 lbf/ft^2)		

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Soil Properties(16) (continued) **Minimum Dynamic Bearing Capacity (continued)** **Firewater Service Complex**			
Soft	440 kPa (9200 lbf/ft^2)	**ESP** No values provided	
Medium Hard	540 kPa (11,300 lbf/ft^2) 670 kPa (14,000 lbf/ft^2)	**Unit 3** 1389 kPa (29,000 lbf/ft^2)	The Unit 3 site characteristic value for minimum dynamic bearing capacity for the FWSC structure is from Table 2.5-215 and falls within (is greater than) the DCD site parameter minimum value for any type of soil: hard, medium, or soft. Based on the equivalent uniform shear wave velocity identified below, the materials beneath the FWSC structure are classified as medium in accordance with Note (7).
Minimum Shear Wave Velocity(8)	300 m/s (1000 ft/s)	**ESP** No value provided	
		Unit 3 Value for each Seismic Category I structure: 2638 m/s (8655 ft/sec) for the reactor building/fuel building 2097 m/s (6880 ft/sec) for the control building 1073 m/s (3520 ft/sec) for the FWSC	The Unit 3 site characteristic value for each Seismic Category I structure is based on the equivalent uniform shear wave velocity over the entire soil column calculated using the formula in Note (8). The value for each structure falls within (is greater than) the DCD site parameter minimum value. As shown in Figures 2.5-229 through 2.5-232, the FB/RB, CB, and FWSC foundations are founded on uniform material. Therefore, the ratio of the largest to the smallest shear wave velocity over each mat foundation level does not exceed 1.7.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject [17]	DCD Site Parameter Value [1] [17]	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Soil Properties [16] (continued)			
Liquefaction Potential			
Seismic Category I structures	None under footprint of Seismic Category I structures resulting from site-specific SSE	**ESP** No value provided	
		Unit 3 None at site-specific SSE under Seismic Category I structures	The Unit 3 site characteristic value for liquefaction falls within (is the same as) the DCD site parameter. As described in Section 2.5.4.8, there is no potential for liquefaction under Unit 3 Seismic Category I structures at the site-specific SSE ground motion. SSAR Table 1.9-1 states that safety-related structures would be founded on rock with no liquefaction potential, or on soil with a factor of safety against liquefaction equal to or greater than 1.1 at the SSE ground motion.
Other than Seismic Category I structures	See Note (14)	See Evaluation column	Note (14) in DCD Table 2.0-1 identifies a requirement to address liquefaction potential under other than Seismic Category I structures. This requirement is not a site parameter. Section 2.5.4.8 provides the results of the liquefaction analysis for the Unit 3 site and addresses potential liquefaction under other than Seismic Category I structures. Seismic Category II structures have no potential for liquefaction. Structures other than Seismic Category I and II structures are located such that a failure of such a structure does not affect the safety of Seismic Category I structures.
Angle of Internal Friction	≥30 degrees	**ESP** No value provided	
		Unit 3 ≥30 degrees	The Unit 3 site characteristic value for angle of internal friction is provided in Section 2.5.4.2.5 and falls within (is the same as) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Seismology			
	SSE Horizontal Ground Response Spectra [9]	See DCD Figure 2.0-1		The DCD site parameter values for SSE response spectra at foundation level are identified as the CSDRS. The CSDRS for the CB and RB/FB are shown in DCD Figure 2.0-1 (horizontal) and in DCD Figure 2.0-2 (vertical). The CSDRS for the FWSC are 1.35 times the accelerations shown in DCD Figure 2.0-1 (horizontal) and in DCD Figure 2.0-2 (vertical) per Note (9) in DCD Table 2.0-1.
	SSE Vertical Ground Response Spectra [9]	See DCD Figure 2.0-2		
			ESP No values provided	
			Unit 3 See Figures 2.5-206, 2.5-207, and 2.5-208	The Unit 3 site characteristic values are identified as the FIRS. The CB FIRS are shown in Figure 2.5-206. The RB/FB FIRS are shown in Figure 2.5-207. The FWSC FIRS are shown in Figure 2.5-208. The comparisons of the DCD site parameter (CSDRS for the CB and RB/FB) and Unit 3 site characteristic values (FIRS for the CB and RB/FB) are provided in Figure 2.0-201 for the horizontal spectra and in Figure 2.0-202 for the vertical spectra. These comparisons demonstrate that the Unit 3 site characteristic values fall within (are less than) the values established by the DCD site parameters. The comparisons of the DCD site parameter (CSDRS for the FWSC) and Unit 3 site characteristic values (FIRS for the FWSC) are provided in Figure 2.0-203 for the horizontal spectra and in Figure 2.0-204 for the vertical spectra. These comparisons demonstrate that the Unit 3 site characteristic values fall within (are less than) the values established by the DCD site parameters.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Hazards in Site Vicinity			
Site Proximity Missiles and Aircraft	< about 10^{-7} per year (for site proximity missile hazards)	**ESP** No value provided	
		Unit 3 No site proximity missile hazards identified	The Unit 3 site characteristic value for site proximity missiles value is that there are no site proximity missile sources identified. As provided in Section 2.2, there are no nearby missile sources identified in the site vicinity and this value falls within (is less than) the DCD site parameter value.
	< about 10^{-7} per year (for aircraft hazards)	**ESP** No value provided	
		Unit 3 Annual aircraft crash probability of 1.07×10^{-7} (includes civil and military aircraft)	The Unit 3 site characteristic value for total probability per year of a civil or military aircraft crashing was estimated per NUREG-0800 as shown in Section 2.2.3.2.2 and the total accident probability falls within (is the same as) the DCD site parameter value.
Volcanic Activity	None	**ESP** No value provided	
		Unit 3 No volcanic activity at the site	The Unit 3 site characteristic value for volcanic activity is that there is no evidence of non-tectonic deformation at the site, such as volcanic intrusion, as presented in SSAR Section 2.5.3.8. The Unit 3 site characteristic value falls within (is the same as) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Hazards in Site Vicinity (continued)			
	Toxic Gases	None*		
	* Maximum toxic gas concentrations at the Main Control Room (MCR) HVAC intakes	<toxicity limits	**ESP** No value provided	
			Unit 3 < toxicity limits	The Unit 3 site characteristic value for toxic gases is that the control room concentration for each chemical analyzed does not exceed the applicable toxicity limit. Based on this result, Seismic Category I Class 1E toxic gas monitoring instrumentation is not required for the MCR HVAC air intakes. The Unit 3 site characteristic value for toxic gases (control room concentrations < toxicity limits) is presented in Section 6.4.5 and falls within (is the same as) the DCD site parameter value for toxic gases (control room concentrations < toxicity limits).
	Required Stability of Slopes [10]			Note (10) in DCD Table 2.0-1 identifies that factors of safety for stability of slopes are not site parameters. These factors are used with slope design features to ensure stability for static and dynamic loading.
	Factor of safety for static (non-seismic) loading	1.5	See Evaluation column	Section 2.5.5.2 specifies that the minimum acceptable long-term static (non-seismic) factor of safety against slope stability failure is 1.5.
	Factor of safety for dynamic (seismic) loading due to site-specific SSE	1.1	See Evaluation column	Section 2.5.5.2 specifies that the minimum acceptable long-term dynamic (seismic) factor of safety against slope stability failure is 1.1.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject [17]	DCD Site Parameter Value [1] [17]	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Maximum Settlement Values for Seismic Category I Buildings [15]			
Maximum Settlement at any corner of basemat			
Under Reactor/Fuel Building	103 mm (4.0 inches)	**ESP** No value provided	
		Unit 3 1 mm (0.05 in) for the maximum settlement of a RB/FB corner	The Unit 3 site characteristic value for the maximum settlement of a corner for the RB/FB foundation is provided in Table 2.5-216 and falls within (is less than) the DCD site parameter value.
Under Control Building	18 mm (0.7 inches)	**ESP** No value provided	
		Unit 3 0.5 mm (0.02 in) for the maximum settlement of a CB corner	The Unit 3 site characteristic value for the maximum settlement of a corner for the CB foundation is provided in Table 2.5-216 and falls within (is less than) the DCD site parameter value.
Under FWSC Structure	17 mm (0.7 inches)	**ESP** No value provided	
		Unit 3 6.6 mm (0.26 in) for the maximum settlement of a FWSC corner	The Unit 3 site characteristic value for the maximum settlement of a corner for the FWSC foundation is provided in Table 2.5-216 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Maximum Settlement Values for Seismic Category I Buildings (continued)			
Averaged Settlement at four corners of basemat			
Under Reactor/Fuel Building	65 mm (2.6 inches)	**ESP** No value provided	
		Unit 3 1 mm (0.05 in) for the maximum settlement of a RB/FB corner	The Unit 3 site characteristic value for the averaged settlement at four corners is the maximum settlement of a corner because each corner settles the same amount, i.e., the maximum amount for a corner. The maximum settlement of a corner for the RB/FB foundation is provided in Table 2.5-216 and falls within (is less than) the DCD site parameter value.
Under Control Building	12 mm (0.5 inches)	**ESP** No value provided	
		Unit 3 0.5 mm (0.02 in) for the maximum settlement of a CB corner	The Unit 3 site characteristic value for the averaged settlement at four corners is the maximum settlement of a corner because each corner settles the same amount, i.e., the maximum amount for a corner. The maximum settlement of a corner for the CB foundation is provided in Table 2.5-216 and falls within (is less than) the DCD site parameter value.
Under FWSC Structure	10 mm (0.4 inches)	**ESP** No value provided	
		Unit 3 6.6 mm (0.26 in) for the maximum settlement of a FWSC corner	The Unit 3 site characteristic value for the averaged settlement at four corners is the maximum settlement of a corner because each corner settles the same amount, i.e., the maximum amount for a corner. The maximum settlement of a corner for the FWSC foundation is provided in Table 2.5-216 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Maximum Settlement Values for Seismic Category I Buildings (continued)			
	Maximum Differential Settlement along the longest mat foundation dimension			
	Within Reactor/Fuel Building	77 mm (3.0 inches)	**ESP** No value provided	
			Unit 3 2 mm (0.07 in)	The Unit 3 site characteristic value for the maximum differential settlement along the longest mat foundation dimension is the maximum settlement of the center of the RB/FB foundation less the maximum settlement for a corner. These values are provided in Table 2.5-216. The difference in these values determines the Unit 3 site characteristic value for the maximum differential settlement for the RB/FB foundation which, as shown, falls within (is less than) the DCD site parameter value.
	Within Control Building	14 mm (0.6 inches)	**ESP** No value provided	
			Unit 3 0.5 mm (0.02 in)	The Unit 3 site characteristic value for the maximum differential settlement along the longest mat foundation dimension is the maximum settlement of the center of the CB foundation less the maximum settlement for a corner. These values are provided in Table 2.5-216. The difference in these values determines the Unit 3 site characteristic value for the maximum differential settlement for the CB foundation which, as shown, falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Maximum Settlement Values for Seismic Category I Buildings (continued) **Maximum Differential Settlement along the longest mat foundation dimension (continued)**			
Under FWSC Structure	12 mm (0.5 inches)	**ESP** No value provided	
		Unit 3 11 mm (0.45 in)	The Unit 3 site characteristic value for the maximum differential settlement along the longest mat foundation dimension is the maximum settlement of the center of the FWSC foundation less the maximum settlement for a corner after the installation of the basemat (applied load excluding weight of basemat). These values are provided in Table 2.5-216. The difference in these values determines the Unit 3 site characteristic value for the maximum differential settlement for the FWSC foundation which, as shown, falls within (is less than) the DCD site parameter value.
Maximum Differential Displacement between Reactor/Fuel Buildings and Control Building			
	85 mm (3.3 inches)	**ESP** No value provided	
		Unit 3 2 mm (0.08 in)	The Unit 3 site characteristic value for the maximum differential displacement between the RB/FB foundation and the CB foundation is the maximum settlement of the center of the RB/FB foundation less the maximum settlement of the center of the CB foundation. For the RB/FB and the CB foundations, the maximum settlement of the center of each is provided in Table 2.5-216. The difference in these values determines the Unit 3 site characteristic value for the maximum differential displacement between the RB/FB foundation and the CB foundation which, as shown, falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) [11]			
	EAB χ/Q			
	0–2 hours	2.00E-03 s/m^3	**ESP and Unit 3** 2.26E-04 s/m^3	The ESP site characteristic value for short-term (accident release) atmospheric dispersion for 0–2 hr χ/Q value at the EAB is defined as the 0–2 hour atmospheric dispersion factor to be used to estimate dose consequences of accidental airborne releases at the EAB. The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Table 2.3-3 and SSAR Table 1.9-1 provide the same value as the ESP. Note that although the EAB location yielding the highest atmospheric dispersion factors was determined by GIS measurement to be 1609 m (1.0 mi) ESE, the SSAR distance of 1416 m (0.88 mi) ESE is conservative and used. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.
	LPZ χ/Q			
	0–8 hours	1.90E-04 s/m^3	**ESP and Unit 3** 2.05E-05 s/m^3	The ESP site characteristic value for short-term (accident release) atmospheric dispersion for 0–8 hr χ/Q value at the LPZ is defined as the 0–8 hour atmospheric dispersion factor to be used to estimate dose consequences of accidental airborne releases at the LPZ. The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Table 2.3-3 and SSAR Table 1.9-1 provide the same value as the ESP. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **LPZ χ/Q (continued)**			
	8–24 hours	1.40E-04 s/m^3	**ESP and Unit 3** 1.36E-05 s/m^3	The ESP site characteristic value for short-term (accident release) atmospheric dispersion for 8–24 hr χ/Q value at the LPZ is defined as the 8–24 hour atmospheric dispersion factor to be used to estimate dose consequences of accidental airborne releases at the LPZ. The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Table 2.3-3 and SSAR Table 1.9-1 provide the same value as the ESP. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.
	1–4 days	7.50E-05 s/m^3	**ESP and Unit 3** 5.58E-06 s/m^3	The ESP site characteristic value for short-term (accident release) atmospheric dispersion for 1–4 day χ/Q value at the LPZ is defined as the 1–4 day atmospheric dispersion factor to be used to estimate dose consequences of accidental airborne releases at the LPZ. The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Table 2.3-3 and SSAR Table 1.9-1 provide the same value as the ESP. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.
	4–30 days	3.00E-05 s/m^3	**ESP and Unit 3** 1.55E-06 s/m^3	The ESP site characteristic value for short-term (accident release) atmospheric dispersion for 4–30 day χ/Q value at the LPZ is defined as the 4–30 day atmospheric dispersion factor to be used to estimate dose consequences of accidental airborne releases at the LPZ. The ESP site characteristic value falls within (is lower than) the DCD site parameter value. SSAR Table 2.3-3 and SSAR Table 1.9-1 provide the same value as the ESP. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued)			
	Control Room χ/Q * * First value is for unfiltered inleakage. Second value is for air intakes (emergency and normal).		Control Room χ/Q values shown on the same row in DCD Table 2.0-1 are in sets below: first a set for unfiltered inleakage, followed by a set for air intakes (emergency and normal).	
	Reactor Building **Unfiltered inleakage**			
	0–2 hours	1.90E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.74E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	2–8 hours	1.30E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.17E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	8–24 hours	5.90E-04 s/m^3	**ESP** No value provided	
			Unit 3 4.07E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Reactor Building (continued)** **Unfiltered inleakage (continued)**			
	1–4 days	5.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 3.42E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	4–30 days	4.40E-04 s/m^3	**ESP** No value provided	
			Unit 3 2.79E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	Air intakes (emergency and normal)			
	0–2 hours	1.50E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.25E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	2–8 hours	1.10E-03 s/m^3	**ESP** No value provided	
			Unit 3 8.88E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) Control Room χ/Q (continued) Reactor Building (continued) Air intakes (emergency and normal) (continued)			
	8–24 hours	5.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 3.41E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	1–4 days	4.20E-04 s/m^3	**ESP** No value provided	
			Unit 3 2.69E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	4–30 days	3.80E-04 s/m^3	**ESP** No value provided	
			Unit 3 2.20E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)**			
	Passive Containment Cooling System/Reactor Building Roof			
	Unfiltered inleakage			
	0–2 hours	3.40E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.58E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
	2–8 hours	2.70E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.34E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
	8–24 hours	1.40E-03 s/m^3	**ESP** No value provided	
			Unit 3 5.61E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
	1–4 days	1.10E-03 s/m^3	**ESP** No value provided	
			Unit 3 3.96E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Passive Containment Cooling System/Reactor Building Roof (continued)** **Unfiltered inleakage (continued)**			
	4–30 days	7.90E-04 s/m^3	**ESP** No value provided	
			Unit 3 3.34E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
	Air intakes (emergency and normal)			
	0–2 hours	3.00E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.31E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
	2–8 hours	2.50E-03 s/m^3	**ESP** No value provided	
			Unit 3 9.35E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
	8–24 hours	1.20E-03 s/m^3	**ESP** No value provided	
			Unit 3 3.72E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Passive Containment Cooling System/Reactor Building Roof (continued)** **Air intakes (emergency and normal) (continued)**			
	1–4 days	9.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 2.70E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
	4–30 days	7.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 2.18E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
	Blowout Panels/Reactor Building Roof **Unfiltered Leakage**			
	0–2 hours	7.00E-03 s/m^3	**ESP** No value provided	
			Unit 3 2.16E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-208 and falls within (is less than) the DCD site parameter value.
	2–8 hours	5.00E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.72E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-208 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Blowout Panels/Reactor Building Roof (continued)** **Unfiltered Leakage (continued)**			
	8–24 hours	2.10E-03 s/m^3	**ESP** No value provided	
			Unit 3 7.21E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-208 and falls within (is less than) the DCD site parameter value.
	1–4 days	1.70E-03 s/m^3	**ESP** No value provided	
			Unit 3 5.25E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-208 and falls within (is less than) the DCD site parameter value.
	4–30 days	1.50E-03 s/m^3	**ESP** No value provided	
			Unit 3 4.20E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-208 and falls within (is less than) the DCD site parameter value.
	Air intakes (emergency and normal)			
	0–2 hours	5.90E-03 s/m^3	**ESP** No value provided	
			Unit 3 2.00E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-208 and falls within (is less than) the DCD site parameter value.
	2–8 hours	4.70E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.38E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-208 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Blowout Panels/Reactor Building Roof (continued)** **Air intakes (emergency and normal) (continued)**			
	8–24 hours	1.50E-03 s/m^3	**ESP** No value provided	
			Unit 3 5.23E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-208 and falls within (is less than) the DCD site parameter value.
	1–4 days	1.10E-03 s/m^3	**ESP** No value provided	
			Unit 3 3.72E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-208 and falls within (is less than) the DCD site parameter value.
	4–30 days	1.00E-03 s/m^3	**ESP** No value provided	
			Unit 3 3.06E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-208 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Turbine Building** **Unfiltered inleakage**			
	0–2 hours	1.20E-03 s/m^3	**ESP** No value provided	
			Unit 3 6.71E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
	2–8 hours	9.80E-04 s/m^3	**ESP** No value provided	
			Unit 3 3.42E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
	8–24 hours	3.90E-04 s/m^3	**ESP** No value provided	
			Unit 3 1.53E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
	1–4 days	3.80E-04 s/m^3	**ESP** No value provided	
			Unit 3 1.17E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Turbine Building (continued)** **Unfiltered inleakage (continued)**			
	4–30 days	3.20E-04 s/m^3	**ESP** No value provided	
			Unit 3 9.19E-05 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
	Air intakes (emergency and normal)			
	0–2 hours	1.20E-03 s/m^3	**ESP** No value provided	
			Unit 3 8.17E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
	2–8 hours	9.80E-04 s/m^3	**ESP** No value provided	
			Unit 3 3.96E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
	8–24 hours	3.90E-04 s/m^3	**ESP** No value provided	
			Unit 3 1.78E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1] [17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Turbine Building (continued)** **Air intakes (emergency and normal) (continued)**			
	1–4 days	3.80E-04 s/m^3	**ESP** No value provided	
			Unit 3 1.50E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
	4–30 days	3.20E-04 s/m^3	**ESP** No value provided	
			Unit 3 1.15E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
	Fuel Building **Unfiltered inleakage**			
	0–2 hours	2.80E-03 s/m^3	**ESP** No value provided	
			Unit 3 2.62E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-204 and falls within (is less than) the DCD site parameter value
	2–8 hours	2.50E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.97E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-204 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Fuel Building (continued)** **Unfiltered inleakage (continued)**			
	8–24 hours	1.25E-03 s/m^3	**ESP** No value provided	
			Unit 3 7.26E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-204 and falls within (is less than) the DCD site parameter value.
	1–4 days	1.10E-03 s/m^3	**ESP** No value provided	
			Unit 3 6.01E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-204 and falls within (is less than) the DCD site parameter value.
	4–30 days	1.00E-03 s/m^3	**ESP** No value provided	
			Unit 3 5.20E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-204 and falls within (is less than) the DCD site parameter value.
	Air intakes (emergency and normal)			
	0–2 hours	2.80E-03 s/m^3	**ESP** No value provided	
			Unit 3 2.15E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-204 and falls within (is less than) the DCD site parameter value
	2–8 hours	2.50E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.59E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-204 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Fuel Building Source (continued)** **Air intakes (emergency and normal) (continued)**			
	8–24 hours	1.25E-03 s/m^3	**ESP** No value provided	
			Unit 3 5.90E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-204 and falls within (is less than) the DCD site parameter value.
	1–4 days	1.10E-03 s/m^3	**ESP** No value provided	
			Unit 3 4.70E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-204 and falls within (is less than) the DCD site parameter value.
	4–30 days	1.00E-03 s/m^3	**ESP** No value provided	
			Unit 3 4.02E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-204 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Radwaste Building** **Unfiltered inleakage**			
				The PCCS vent χ/Q values are assumed to bound the χ/Q values for any release from the RW Building based on distance and direction to the CR receptors, and the PCCS vent χ/Q values are used to evaluate releases from the RW Building in the DCD (Section 15.3.16). The PCCS χ/Q values are compared to the RW Building χ/Q results.
	0–2 hours	3.40E-03 s/m^3	**ESP** No value provided	
			Unit 3 6.13E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-207 and falls within (is less than) the DCD site parameter value.
	2–8 hours	2.70E-03 s/m^3	**ESP** No value provided	
			Unit 3 4.90E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-207 and falls within (is less than) the DCD site parameter value.
	8–24 hours	1.40E-03 s/m^3	**ESP** No value provided	
			Unit 3 2.19E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-207 and falls within (is less than) the DCD site parameter value.
	1–4 days	1.10E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.58E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-207 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Radwaste Building (continued)** **Unfiltered inleakage (continued)**			
	4–30 days	7.90E-04 s/m^3	**ESP** No value provided	
			Unit 3 1.29E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-207 and falls within (is less than) the DCD site parameter value.
	Air intakes (emergency and normal)			
	0–2 hours	3.00E-03 s/m^3	**ESP** No value provided	
			Unit 3 4.69E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-207 and falls within (is less than) the DCD site parameter value.
	2–8 hours	2.50E-03 s/m^3	**ESP** No value provided	
			Unit 3 3.76E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-207 and falls within (is less than) the DCD site parameter value.
	8–24 hours	1.20E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.66E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-207 and falls within (is less than) the DCD site parameter value.
	1–4 days	9.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 1.17E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-207 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Control Room χ/Q (continued)** **Radwaste Building (continued)** **Air intakes (emergency and normal) (continued)**			
	4–30 days	7.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 9.96E-05 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-207 and falls within (is less than) the DCD site parameter value.
	Technical Support Center χ/Q		The Technical Support Center χ/Q values shown on the same row in DCD Table 2.0-1 for unfiltered inleakage and for the air intakes (emergency and normal) were assumed to be the same, therefore, one comparison for each set of TSC χ/Q values is provided below.	
	Reactor Building **TSC Unfiltered inleakage and TSC Air intakes (emergency and normal)**			
	0–2 hours	1.00E-03 s/m^3	**ESP** No value provided	
			Unit 3 2.63E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	2–8 hours	6.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 2.17E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	8–24 hours	3.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 9.35E-05 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Reactor Building (continued)** **TSC Unfiltered inleakage and TSC Air intakes (emergency and normal) (continued)**			
	1–4 days	2.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 6.71E-05 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	4–30 days	1.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 5.21E-05 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-201 and falls within (is less than) the DCD site parameter value.
	Turbine Building **TSC Unfiltered inleakage and TSC Air intakes (emergency and normal)**			
	0–2 hours	2.00E-03 s/m^3	**ESP** No value provided	
			Unit 3 2.00E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is the same as) the DCD site parameter value.
	2–8 hours	1.50E-03 s/m^3	**ESP** No value provided	
			Unit 3 1.13E-03 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Meteorological Dispersion (χ/Q) (continued) **Turbine Building (continued)** **TSC Unfiltered inleakage and TSC Air intakes (emergency and normal) (continued)**			
8–24 hours	8.00E-04 s/m^3	**ESP** No value provided	
		Unit 3 4.45E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
1–4 days	6.00E-04 s/m^3	**ESP** No value provided	
		Unit 3 3.78E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
4–30 days	5.00E-04 s/m^3	**ESP** No value provided	
		Unit 3 3.27E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-202 and falls within (is less than) the DCD site parameter value.
Passive Containment Cooling System/Reactor Building Roof **TSC Unfiltered inleakage and TSC Air intakes (emergency and normal)**			
0–2 hours	2.00E-03 s/m^3	**ESP** No value provided	
		Unit 3 4.40E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
2–8 hours	1.10E-03 s/m^3	**ESP** No value provided	
		Unit 3 3.64E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Meteorological Dispersion (χ/Q) (continued) **Passive Containment Cooling System/Reactor Building Roof (continued)** **TSC Unfiltered inleakage and TSC Air intakes (emergency and normal) (continued)**			
	8–24 hours	5.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 1.52E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
	1–4 days	4.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 1.16E-04 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.
	4–30 days	3.00E-04 s/m^3	**ESP** No value provided	
			Unit 3 8.78E-05 s/m^3	The Unit 3 site characteristic value is provided in Table 2.3-203 and falls within (is less than) the DCD site parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Long Term Dispersion Estimates [12]			
NAPS COL 12.2-2-A	χ/Q: Reactor/Fuel Building Ventilation Stack (RB-VS) Turbine Building Ventilation Stack (TB-VS) Radwaste Building Ventilation Stack (RW-VS) D/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3 1.0E-08 m^{-2} 6.0E-09 m^{-2} 3.0E-08 m^{-2}	**ESP** The ESP site characteristic values for long term (routine release) atmospheric dispersion estimates are based on the maximally exposed individual (MEI) for each pathway. **Unit 3** The Unit 3 site characteristic values assume conservatively, that each sensitive receptor (meat animal, vegetable garden, residence) is at the location of the closest receptor.	The ESP site characteristic values for long term (routine release) atmospheric dispersion estimates are defined based on type of sensitive receptor (MEI) and decay time. Each of these values is compared with the appropriate DCD site parameter values, χ/Q or D/Q, below. Each ESP site characteristic value that is equal to or less than a DCD site parameter value results in a lower estimated dose for the same source term, and conversely, a higher χ/Q or D/Q results in a higher estimated dose. As shown below, every ESP site characteristic value does not fall within (some are greater than) the DCD site parameter value. As shown further below, every Unit 3 site characteristic value also does not fall within (some are greater than) the DCD site parameter value. Per Note (12) of DCD Table 2.0-1, if a site-specific χ/Q value exceeds the site parameter value, the release concentrations in DCD Table 12.2-17 must be adjusted proportionate to the change in χ/Q using the stack release information in DCD Table 12.2-16, which is replaced by the Unit 3 release information in Table 2.3-16R, to show the 10 CFR 20 limits are met; and the annual average doses in DCD Table 12.2-18b must be changed to show the 10 CFR 50, Appendix I limits are met. Per DCD COL Item 12.2-2-A, calculation bases in DCD Tables 12.2-15 and 12.2-18a are replaced with site-specific values for calculation of airborne concentrations and doses. Tables 12.2-15R and 12.2-18bR identify the replacement of DCD information. This table identifies that there are Unit 3 site characteristic values that do not fall within (are greater than) the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results.

(continued)

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS COL 2.0-1-A

Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
Part 1 – Evaluation of DCD Site Parameters			
Long Term Dispersion Estimates [12] (continued)			
			Because the site characteristic values for both the ESP and Unit 3 are defined based on releases from the plant parameter envelope as shown in Figure 2.0-205, there is a single χ/Q and D/Q value for each type of sensitive receptor (MEI) and decay time, rather than values for releases from each ventilation stack. Each site characteristic χ/Q value is compared with all three DCD site parameter χ/Q values, which correspond to a value for each of the three buildings with a ventilation stack. Each site characteristic D/Q value is similarly compared with all three DCD site parameter D/Q values.
χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP and Unit 3** 3.7×10^{-6} s/m^3, annual average, undepleted/no decay, EAB, east-southeast, 1.4 km (0.88 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average EAB undepleted/no decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value is provided in Table 2.3-16R and falls within (is the same as) the ESP site characteristic value.
χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP and Unit 3** 3.7×10^{-6} s/m^3, annual average, undepleted/2.26-day decay, EAB, east-southeast, 1.4 km (0.88 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average EAB undepleted/2.26-day decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value is provided in Table 2.3-16R and falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Long Term Dispersion Estimates[12] (continued)			
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP and Unit 3** 3.3×10^{-6} s/m^3, annual average, depleted/8.00-day decay, EAB, east-southeast, 1.4 km (0.88 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average EAB depleted/8.00-day decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value is provided in Table 2.3-16R and falls within (is the same as) the ESP site characteristic value.
	D/Q: RB-VS TB-VS RW-VS	 1.0E-08 m^{-2} 6.0E-09 m^{-2} 3.0E-08 m^{-2}	**ESP and Unit 3** 1.2×10^{-8} 1/m^2, annual average, D/Q value, EAB, east-southeast*, 1.4 km (0.88 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average EAB D/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value is provided in Table 2.3-16R and falls within (is the same as) the ESP site characteristic value. * The direction is south and the distance is 1 km (0.62 mi) as shown in ESP-ER Table 2.7-16 and in Table 2.3-16R.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Long Term Dispersion Estimates [12] (continued)			
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP** 2.4×10^{-6} s/m^3, annual average, undepleted/no decay, nearest resident, north-northeast, 1.5 km (0.96 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average resident undepleted/no decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values.
NAPS ESP VAR 2.0-1a			**Unit 3** 4.2×10^{-6} s/m^3 east-southeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) the ESP site characteristic value.
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP** 2.4×10^{-6} s/m^3, annual average, undepleted/2.26-day decay, nearest resident, north-northeast, 1.5 km (0.96 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average resident undepleted/2.26 day decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values.
NAPS ESP VAR 2.0-1b			**Unit 3** 4.1×10^{-6} s/m^3 east-southeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. This Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Long Term Dispersion Estimates [12] (continued)			
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP** 2.1 × 10^{-6} s/m^3, annual average, depleted/8.00-day decay, nearest resident, north-northeast, 1.5 km (0.96 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average resident depleted/8.00-day decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values.
NAPS ESP VAR 2.0-1c			**Unit 3** 3.7 × 10^{-6} s/m^3 east-southeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) the ESP site characteristic value.
	D/Q: RB-VS TB-VS RW-VS	 1.0E-08 m^{-2} 6.0E-09 m^{-2} 3.0E-08 m^{-2}	**ESP** 7.2 × 10^{-9} 1/m^2, annual average, nearest resident, north-northeast, 1.5 km (0.96 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average resident D/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values.
NAPS ESP VAR 2.0-1d			**Unit 3** 1.1 × 10^{-8} 1/m^2 north-northeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) two of the ESP site characteristic values.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Long Term Dispersion Estimates[12] (continued)			
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP** 1.4×10^{-6} s/m^3, annual average, undepleted/ no decay, nearest meat animal, southeast, 2.2 km (1.37 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average meat animal undepleted/no decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values.
NAPS ESP VAR 2.0-1e			**Unit 3** 4.2×10^{-6} s/m^3 east-southeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) the ESP site characteristic value.
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP** 1.4×10^{-6} s/m^3, annual average, undepleted/2.26-day decay, nearest meat animal, southeast, 2.2 km (1.37 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average meat animal undepleted/2.26-day decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. This ESP site characteristic value is 1.4×10^{-6} s/m^3 and does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results.
NAPS ESP VAR 2.0-1f			**Unit 3** 4.1×10^{-6} s/m^3 east-southeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Long Term Dispersion Estimates [12] (continued)			
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP** 1.2 × 10^{-6} s/m^3, annual average, depleted/8.00-day decay, nearest meat animal, southeast, 2.2 km (1.37 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average meat animal depleted/8.00-day decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values.
NAPS ESP VAR 2.0-1g			**Unit 3** 3.7 × 10^{-6} s/m^3 east-southeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) the ESP site characteristic values.
	D/Q: RB-VS TB-VS RW-VS	 1.0E-08 m^{-2} 6.0E-09 m^{-2} 3.0E-08 m^{-2}	**ESP** 3.1 × 10^{-9} 1/m^2, annual average, nearest meat animal, southeast, 2.2 km (1.37 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average meat animal D/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value falls within (is smaller than) the DCD site parameter values.
NAPS ESP VAR 2.0-1h			**Unit 3** 1.1 × 10^{-8} 1/m^2 north-northeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) two of the ESP site characteristic values.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Long Term Dispersion Estimates[12] (continued)			
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP** 2.0×10^{-6} s/m^3, annual average, undepleted/no decay, nearest vegetable garden, northeast, 1.5 km (0.94 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average vegetable garden undepleted/no decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values.
NAPS ESP VAR 2.0-1i			**Unit 3** 4.2×10^{-6} s/m^3 east-southeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) the ESP site characteristic value.
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP** 2.0×10^{-6} s/m^3, annual average, undepleted/2.26-day decay, nearest vegetable garden, northeast, 1.5 km (0.94 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average vegetable garden undepleted 2.26-day decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values.
NAPS ESP VAR 2.0-1j			**Unit 3** 4.1×10^{-6} s/m^3 east-southeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Long Term Dispersion Estimates[12] (continued)			
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP** 1.8×10^{-6} s/m^3, annual average, depleted/8.00-day decay, nearest vegetable garden, northeast, 1.5 km (0.94 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average vegetable garden depleted/8.00-day decay χ/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value falls within (is less than) two of the DCD site parameter values.
NAPS ESP VAR 2.0-1k			**Unit 3** 3.7×10^{-6} s/m^3 east-southeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) the ESP site characteristic value.
	D/Q: RB-VS TB-VS RW-VS	 1.0E-08 m^{-2} 6.0E-09 m^{-2} 3.0E-08 m^{-2}	**ESP** 6.0×10^{-9} 1/m^2, annual average, nearest vegetable garden, northeast, 1.5 km (0.94 mi)	The ESP site characteristic value for this long term dispersion estimate is defined as the maximum annual average vegetable garden D/Q value for use in determining gaseous pathway doses to the maximally exposed individual. The ESP site characteristic value does not fall within (is greater than) two of the DCD site parameter values.
NAPS ESP VAR 2.0-1l			**Unit 3** 1.1×10^{-8} 1/m^2 north-northeast, 1.2 km (0.74 mi)	The Unit 3 site characteristic value for this long term dispersion estimate is provided in Table 2.3-16R. The Unit 3 site characteristic value does not fall within (is greater than) two of the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value does not fall within (is greater than) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS COL 2.0-1-A	**Part 1 – Evaluation of DCD Site Parameters**			
	Long Term Dispersion Estimates[12] (continued)			
	χ/Q: RB-VS TB-VS RW-VS	 3.0E-07 s/m^3 2.0E-07 s/m^3 2.0E-05 s/m^3	**ESP and Unit 3** No value provided for annual average, nearest cow-milk, undepleted/no decay χ/Q value; annual average undepleted/2.26-day decay χ/Q value; and annual average depleted/8.00-day decay	The ESP and Unit 3 site characteristic values for each of these long term χ/Q dispersion coefficients is "No value provided." The milk exposure pathway was not considered because there are no reported cows or goats used for milk production in the near vicinity of the site, within 5 miles. Each ESP and Unit 3 site characteristic value falls within (is smaller than) the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic values fall within (are the same as) the ESP site characteristic value.
	D/Q: RB-VS TB-VS RW-VS	 1.0E-08 m^{-2} 6.0E-09 m^{-2} 3.0E-08 m^{-2}	**ESP and Unit 3** No value provided for annual average, nearest cow-milk	The ESP and Unit 3 site characteristic values for this long term D/Q dispersion estimate is "No value provided." The milk exposure pathway was not considered because there are no reported cows or goats used for milk production in the near vicinity of the site, within 5 miles. The ESP and Unit 3 site characteristic values fall within (are the smaller than) the DCD site parameter values. See Section 12.2 for the site-specific concentration and dose analysis inputs and results. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-1

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter			
Site Characteristic			
Exclusion Area Boundary	No value provided	**ESP** Perimeter of a 1524 m (5000 ft) radius circle from the center of the abandoned Unit 3 containment **Unit 3** 10 CFR 100.21(a) Meets requirement	The ESP site characteristic value is defined as the area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. The Unit 3 site characteristic is presented as a criterion and the value is described in SSAR Table 1.9-1 as: "The exclusion area boundary is the perimeter of a 5000-ft-radius circle from the center of the abandoned Unit 3 containment." The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.
Low Population Zone	No value provided	**ESP** 9.7 km (6 mi) radius circle centered at the Unit 1 containment building. **Unit 3** 10 CFR 100.21(a) Meets requirement	The ESP site characteristic value is defined as the area immediately surrounding the exclusion area which contains residents. The Unit 3 site characteristic is presented as a criterion and the value is described in SSAR Table 1.9-1 as: "The low population zone is a 6-mile radius circle centered at the Unit 1 containment building." The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-1

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter			
Population Center Distance	No value provided	**ESP** Minimum of 12.9 km (8 mi) **Unit 3** 10 CFR 100.21(b) Meets requirement	The ESP site characteristic value is defined as the minimum allowable distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents. The Unit 3 site characteristic is presented as a criterion and the value is described in SSAR Table 1.9-1 as: "The distance from the ESP plant parameter envelope to the nearest boundary of a densely populated center containing more than about 25,000 residents is not less than one and one-third times the distance from the ESP plant parameter envelope to the outer boundary of the LPZ." The Unit 3 site characteristic criterion equates to a minimum of 12.9 km (8 mi) because the Unit 3 LPZ is a 9.7 km (6 mi) radius circle. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value. Unit 3 meets this criterion because, as stated in SSAR Section 2.1.3.5, the nearest population center to Unit 3 with more than 25,000 residents is the City of Charlottesville and the closest point of this city to Unit 3 is 36 miles west.
Maximum Dry-Bulb Temperature 100-year return period	No value provided	**ESP and Unit 3** 42.8ºC (109ºF)	The ESP site characteristic value is defined as the ambient dry-bulb temperature that has a 1% annual probability of being exceeded (100-year mean recurrence interval). The Unit 3 site characteristic value is provided in SSAR Table 2.3-18 and SSAR Table 1.9-1; and falls within (is the same as) the ESP site characteristic value.
Minimum Dry-Bulb Temperature 99.6% annual exceedance	No value provided	**ESP and Unit 3** -10ºC (14ºF)	The ESP site characteristic value is defined as the ambient dry-bulb temperature below which dry-bulb temperature will fall 0.4% of the time annually. The Unit 3 site characteristic value is provided as the 0.4% annual exceedance value for minimum dry bulb temperature in SSAR Table 2.3-18 and SSAR Table 1.9-1; and falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-1

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter			
Minimum Dry-Bulb Temperature (continued)			
100-year return period	No value provided	**ESP and Unit 3** –28.3ºC (–19ºF)	The ESP site characteristic value is defined as the ambient dry-bulb temperature for which a 1% annual probability of a lower dry-bulb temperature exists (100-year mean recurrence interval). The Unit 3 site characteristic value is provided in SSAR Tables 2.3-18 and 1.9-1, and falls within (is the same as) the ESP site characteristic value.
Maximum Wet-Bulb Temperature 100-year return period	No value provided	**ESP and Unit 3** 31.1ºC (88ºF)	The ESP site characteristic value is defined as the ambient wet-bulb temperature that has a 1 percent annual probability of being exceeded (100-year mean recurrence interval). The Unit 3 site characteristic value is provided in SSAR Table 2.3-18 and SSAR Table 1.9-1; and falls within (is the same as) the ESP site characteristic value.
Ultimate Heat Sink Ambient Air Temperature and Humidity			Although the Unit 3 site characteristic value is presented for comparison with the ESP site characteristic value, the ultimate heat sink (UHS) for the passive Unit 3 ESBWR design does not use safety-related engineered underground reservoirs or storage basins. Comparisons of meteorological conditions are provided as information required per 10 CFR 52.79(b)(1).
Meteorological Conditions Resulting in the Minimum Water Cooling During Any 1 Day	No value provided	**ESP and Unit 3** 26.1ºC (78.9ºF) wet-bulb temperature with coincident 30.9ºC (87.7ºF) dry-bulb temperature	The ESP site characteristic value is defined as the historic worst 1-day daily average of wet-bulb temperatures and coincident dry-bulb temperatures. The Unit 3 site characteristic value is provided in SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.8, and falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-1

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter			
Ultimate Heat Sink Ambient Air Temperature and Humidity (continued)			
Meteorological Conditions Resulting in the Minimum Water Cooling During Any Consecutive 5 days	No value provided	**ESP and Unit 3** 25.3ºC (77.6ºF) wet-bulb temperature with coincident 27.2ºC (80.9ºF) dry-bulb temperature	The ESP site characteristic value is defined as the historic worst 5-day daily average of wet-bulb temperatures and coincident dry-bulb temperatures resulting in minimum water cooling. The Unit 3 site characteristic value is provided in SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.8, and falls within (is the same as) the ESP site characteristic value.
Meteorological Conditions Resulting in the Maximum Evaporation and Drift Loss During Any Consecutive 30 Days	No value provided	**ESP and Unit 3** 24.6ºC (76.3ºF) wet-bulb temperature with coincident 26.4ºC (79.5ºF) dry-bulb temperature	The ESP site characteristic value is defined as the historic worst 30-day daily average of wet-bulb temperatures and coincident dry-bulb temperatures. The Unit 3 site characteristic value is provided in SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.8, and falls within (is the same as) the ESP site characteristic value.
Meteorological Conditions Resulting in the Maximum Water Freezing in the UHS Water Storage Facility	No value provided	**ESP and Unit 3** 179 degree(C)-days (322 degree(F)-days) below freezing	The ESP site characteristic value is defined as the historic maximum cumulative degree-days below freezing. The Unit 3 site characteristic value is provided in SSAR Table 1.9-1, which refers to SSAR Section 2.3.1.3.8, and falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-1

Subject(17)	DCD Site Parameter Value(1)(17)	Site Characteristic	Evaluation
Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter			
Hydrology			
Proposed Facility Boundaries	No value provided	**ESP** Proposed facility boundary as shown in ESP, Appendix A, Figure 1. (Reference 2.0-203) Figure 1 shows the proposed facility boundary using the boundary corners numbered 1-8. Notes 1 and 2 apply.	The ESP site characteristic value is defined as the ESP site boundary map. The Unit 3 site characteristic value, as shown in Figure 2.0-205, falls within (power block buildings which could have postulated accidental fission product releases are located within) the ESP site characteristic value.
Proposed Facility Boundaries	No value provided	**Unit 3** Figure 2.0-205, which shows that the Unit 3 power block buildings which could have postulated accidental fission product releases are located within the Figure 1 proposed facility boundary.	

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-1	**Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter**			
	Hydrology (continued)			
NAPS ESP VAR 2.0-7a			Coordinates of the proposed facility boundaries are shown in Figure 2.0-205.	ESP, Appendix A, Figure 1, Note 1 states: "North Anna Site and State NAD 83 (South Zone) coordinates are shown as noted." There are two sets of values given as Coordinates (NAPS GRID) and Coordinates (State NAD 83 South Zone). The Unit 3 site characteristics are two sets of values given in Figure 2.0-205 as COORDINATES (NAPS U1 & U2 GRID) and COORDINATES (STATE PLANE NAD 83 VA SOUTH ZONE). The Unit 3 values for the COORDINATES (NAPS U1 & U2 GRID) fall within (are the same as) the ESP Coordinates (NAPS GRID) values. The Unit 3 values for the COORDINATES (STATE PLANE NAD 83 VA SOUTH ZONE) do not fall within (are different from) the ESP Coordinates (State NAD 83 South Zone) values.
NAPS ESP VAR 2.0-7b			No removal of abandoned mat foundations unless a Unit 3 Seismic Category I or II structure would be located above a foundation.	ESP, Appendix A, Figure 1, Note 2 states: "Abandoned Unit 3 and 4 Reactor Building Mat Foundations are to be removed." The Unit 3 Site characteristic is no removal of abandoned mat foundations unless a Unit 3 Seismic Category I or II structure would be located above a foundation. The Unit 3 site characteristic does not fall within (is not the same as) the ESP site characteristic.
	Minimum Lake Water Level	No value provided	**ESP and Unit 3** 242 ft msl	The ESP site characteristic value is defined as the low water surface shutdown elevation for operation of NAPS Units 1 and 2, and Unit 3. The Unit 3 site characteristic value is provided in Section 2.4.14 and falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject[17]	DCD Site Parameter Value[1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-1	**Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter**			
	Hydrology (continued)			
	Frazil and Anchor Ice	No value provided	**ESP and Unit 3** Potential for formation of frazil and anchor ice	The ESP site characteristic value is defined as the accumulated ice formation in a turbulent flow condition. The Unit 3 site characteristic value is provided in SSAR Table 1.9-1, which refers to SSAR Section 2.4.7.4, and falls within (is the same as) the ESP site characteristic value.
	Maximum Ice Thickness	No value provided	**ESP and Unit 3** 43.4 cm (17.1 in) thick	The ESP site characteristic value is defined as the ice sheet thickness at Lake Anna (based on maximum cumulative degree-days below freezing of 178.8ºC (321.8ºF)). The Unit 3 site characteristic value is provided in SSAR Section 2.4.7 and falls within (is the same as) the ESP site characteristic value.
	Max Cumulative Degree-Days Below Freezing	No value provided	**ESP** 178.8 degree(C)-days (321.8 degree(F)-days)	The ESP site characteristic value is defined as the measure of severity of winter weather conditions conducive to ice formation (computed using air temperature data from the Piedmont Research Station). The Unit 3 site characteristic value is provided in SSAR Section 2.3.1.3.8 and falls within (is greater than–essentially the same as) the ESP site characteristic value.
			Unit 3 179 degree(C)-days (322 degree(F)-days)	
	Hydraulic Conductivity	No value provided	**ESP** 1.0 m/d (3.4 ft/d)	The ESP site characteristic value is defined as the groundwater flow rate per unit hydraulic gradient. SSAR Table 1.9-1 identifies the hydraulic conductivity as 1.0 m/d (3.4 ft/d).
NAPS ESP VAR 2.0-2			**Unit 3** 3.0 m/d (9.9 ft/d)	The Unit 3 site characteristic value is provided in Section 2.4.12 and does not fall within (is greater than) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
NAPS SUP 2.0-1	**Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter**			
	Hydrology (continued)			
	Hydraulic Gradient	No value provided	**ESP** 0.03 m/m (0.03 ft/ft)	The ESP site characteristic value is defined as the slope of groundwater surface under unconfined conditions or slope of hydraulic pressure head under confined conditions. SSAR Table 1.9-1 identifies the hydraulic gradient as 0.03 m/m (0.03 ft/ft).
NAPS ESP VAR 2.0-3			**Unit 3** 0.04 m/m (0.04 ft/ft)	The Unit 3 site characteristic value is provided in Section 2.4.12 and does not fall within (is greater than) the ESP site characteristic value.
	Basic Geologic and Seismic Information Capable Tectonic Structures	No value provided	**ESP and Unit 3** No fault displacement potential within the investigative area	The ESP site characteristic value is defined as no fault displacement potential within the investigative area. The Unit 3 site characteristic value is provided in SSAR Sections 2.5.1.2.4 and 2.5.3.2.2, as identified in SSAR Table 1.9-1. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-1	**Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter**			
	Vibratory Ground Motion Design Response Spectra	No value provided	**ESP** ESP, Appendix A, Figure 2	The ESP site characteristic values are the horizontal and vertical response spectra provided in ESP, Appendix A, Figure 2. SSAR Table 1.9-1 states that the site-specific response spectra are provided in SSAR Section 2.5.2.6. That section includes SSAR Figure 2.5-48A which is the same as ESP, Appendix A, Figure 2.
NAPS ESP VAR 2.0-4			**Unit 3** Figure 2.5-205	The Unit 3 site characteristic values are the horizontal and vertical response spectra provided in Figure 2.5-205. The Unit 3 site characteristic values (response spectra) do not fall within (are not lower than) the ESP site characteristic values (response spectra) at every frequency. Figure 2.0-206 and Table 2.0-202 compare the ESP and Unit 3 horizontal response spectra. Figure 2.0-207 and Table 2.0-203 compare the ESP and Unit 3 vertical response spectra. While the figures are essentially overlapping curves at low frequencies, the tables show where the Unit 3 spectra exceed the ESP spectra.
	Stability of Subsurface Materials and Foundations **Zone III Weathered Rock (205 ft–298 ft)**			
	Minimum Bearing Capacity	No value provided	**ESP** 766 kPa (16 ksf) **Unit 3** 958 kPa (20 ksf)	The ESP site characteristic value is defined as the allowable load-bearing capacity of layer supporting plant structures. The Unit 3 site characteristic value is provided in Table 2.5-215 and falls within (is greater than) the ESP site characteristic value. SSAR Table 1.9-1 refers to the value in SSAR Table 2.5-47, which is 766 kPa (16 ksf).

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-1

Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter			
Stability of Subsurface Materials and Foundations (continued) **Zone III Weathered Rock (205 ft–298 ft) (continued)**			
Minimum Shear Wave Velocity	No value provided	**ESP** 610 m/sec (2000 ft/sec) **Unit 3** 914 m/sec (3000 ft/sec)	The ESP site characteristic value is defined as the propagation of shear waves through foundation materials. The Unit 3 site characteristic value is the best estimate shear wave velocity in Table 2.5-212. This corresponds to the best estimate ESP shear wave velocity in SSAR Table 1.9-1, which refers to SSAR Table 2.5-45, and FSER Section 2.5.4.1.7 (Reference 2.0-202). The Unit 3 site characteristic value falls within (is greater than) the ESP site characteristic value.
Zone III–IV			
Minimum Bearing Capacity	No value provided	**ESP and Unit 3** 3830 kPa (80 ksf)	The ESP site characteristic value is defined as the allowable load-bearing capacity of layer supporting plant structures. The Unit 3 site characteristic value is provided in Table 2.5-215 falls within (is the same as) the ESP site characteristic value. SSAR Table 1.9-1 refers to the value in SSAR Table 2.5-47, which is 3830 kPa (80 ksf).
Minimum Shear Wave Velocity	No value provided	**ESP** 1006 m/sec (3300 ft/sec) **Unit 3** 1372 m/sec (4500 ft/sec)	The ESP site characteristic value is defined as the propagation of shear waves through foundation materials. The Unit 3 site characteristic value is the best estimate shear wave velocity in Table 2.5-212. This corresponds to the best estimate ESP shear wave velocity in SSAR Table 1.9-1, which refers to SSAR Table 2.5-45, and FSER Section 2.5.4.1.7 (Reference 2.0-202). The Unit 3 site characteristic value falls within (is greater than) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-1

Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter			
Stability of Subsurface Materials and Foundations (continued)			
Zone IV Bedrock (188ft–298ft)			
Minimum Bearing Capacity	No value provided	**ESP and Unit 3** 7661 kPa (160 ksf)	The ESP site characteristic value is defined as the allowable load-bearing capacity of layer supporting plant structures. The Unit 3 site characteristic value falls within (is the same as) the ESP site characteristic value. Minimum bearing capacities are provided in Table 2.5-215. SSAR Table 1.9-1 refers to the value in SSAR Table 2.5-47, which is 7661 kPa (160 ksf).
Minimum Shear Wave Velocity	No value provided	**ESP** 1920 m/sec (6300 ft/sec) **Unit 3** 2743 m/sec (9000 ft/s)	The ESP site characteristic value is defined as the propagation of shear waves through foundation materials. The Unit 3 site characteristic value is the best estimate shear wave velocity in Table 2.5-212. This corresponds to the best estimate ESP shear wave velocity in SSAR Table 1.9-1, which refers to SSAR Table 2.5-45, and FSER Section 2.5.4.1.7 (Reference 2.0-202). The Unit 3 site characteristic value falls within (is greater than) the ESP site characteristic value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-1	**Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter**			
	Bounding Parameters		In the rows below, this column contains ESP Design Parameters and Unit 3 Design Characteristics	
	Maximum Cooling Water Flow Rate – Unit 3	No value provided	**ESP Table B-1 and Unit 3** 5056.3 m^3/h (49.6 cfs)	The ESP bounding design parameter value is defined as the maximum instantaneous withdrawal rate from the North Anna reservoir. The Unit 3 design characteristic value is provided in SSAR Section 2.4.1 and falls within (is the same as) the ESP bounding design parameter value.
	Minimum Site Grade	No value provided	**ESP, Table B-1** 82.6 m (271 ft) msl **Unit 3** 88.4 m (290 ft) msl	The ESP bounding design parameter value is defined as the finished site grade. The Unit 3 design characteristic value is provided in Figure 2.1-201 and falls within (is greater than) the ESP bounding design parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1] [17]	Site Characteristic	Evaluation
NAPS SUP 2.0-1	**Part 2 – Evaluation of ESP Site Characteristics and Design Parameters For Which There is No Corresponding DCD Site Parameter**			
NAPS ESP VAR 2.0-6	**Source Term**			
	Gaseous (Post Accident)	See Evaluation column	**ESP** Values in ESP Appendix B tables **SSAR Table 1.9-1** Values in SSAR Section 15.4 tables (maximum values) **Unit 3** Values in DCD Section 15.4 tables	ESP (design) controlling parameters superseded. Design basis accident (DBA) analyses evaluated in SSAR Chapter 15 were based on accidents and associated source terms for the AP1000, ABWR, and the ESBWR plant designs. The source terms for the DBAs evaluated for the ESBWR in DCD Chapter 15 are not bounded by the ESP source terms (included in ESP-003, Appendix B) in all cases. This is variance NAPS ESP VAR 2.0-6. Calculated doses are shown in DCD Chapter 15 to be within limits set by regulatory guidance documents and applicable regulations. Unit 3 site-specific short term (accident) meteorological dispersion values (χ/Q) are demonstrated in Part 1 of this table to fall within the associated DCD site parameter values. Therefore, the doses for the accidents evaluated in DCD Chapter 15 are bounding for Unit 3 and are within limits set by regulatory guidance documents and applicable regulations.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-2	**Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value**			
	Winter Precipitation			
	100-year Snowpack plus 48-hour Maximum Snowfall	No value provided	**SSAR Table 1.9-1** 2.18 kPa (45.5 lb/sq ft) **Unit 3** See the DCD site parameter "Precipitation (for Roof Design), Maximum Roof Load" under Part 1 of this table.	SSAR Table 1.9-1 specifies a value of 2.18 kPa (45.5 lb/sq ft) as the 48-hour maximum snowfall (72.4 cm (28.5 inches), at 0.72 kPa (15 lb/sq ft)) on top of a 100-year return snowpack (1.46 kPa (30.5 lb/sq ft)).
	Distribution Coefficients (K_d)			
	Mn-54	No value provided	**SSAR Table 1.9-1** 50 cm^3/g	The SSAR Table 1.9-1 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is consistent with SSAR Table 2.4-20.
NAPS ESP VAR 2.0-5a			**Unit 3** 8.37 cm^3/g	The Unit 3 site characteristic value listed in Table 2.4-208 (10% K_d) does not fall within (is less than) the SSAR site characteristic value. Measured K_d values are presented in Table 2.4-207 and show that the Unit 3 site characteristic value is conservative. See Section 2.4.13 for the radionuclide transport analysis.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-2	**Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value**			
	Distribution Coefficients (K_d) (continued)			
	Fe-55	No value provided	**SSAR Table 1.9-1** 165 cm^3/g	The SSAR Table 1.9-1 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is consistent with SSAR Table 2.4-20.
NAPS ESP VAR 2.0-5b			**Unit 3** 6.81 cm^3/g	The Unit 3 site characteristic value listed in Table 2.4-208 (10% K_d) does not fall within (is less than) the SSAR site characteristic value. Measured K_d values are presented in Table 2.4-207 and show that the Unit 3 site characteristic value is conservative. See Section 2.4.13 for the radionuclide transport analysis.
	Co-60	No value provided	**SSAR Table 1.9-1** 60 cm^3/g	The SSAR Table 1.9-1 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is consistent with SSAR Table 2.4-20.
NAPS ESP VAR 2.0-5c			**Unit 3** 9.19 cm^3/g	The Unit 3 site characteristic value listed in Table 2.4-208 (10% K_d) does not fall within (is less than) the SSAR site characteristic value. Measured K_d values are presented in Table 2.4-207 and show that the Unit 3 site characteristic value is conservative. See Section 2.4.13 for the radionuclide transport analysis.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-2	**Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value**			
	Distribution Coefficients (K_d) (continued)			
	Zn-65	No value provided	**SSAR Table 1.9-1** 200 cm^3/g	The SSAR Table 1.9-1 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is consistent with SSAR Table 2.4-20.
NAPS ESP VAR 2.0-5d			**Unit 3** 3.63 cm^3/g	The Unit 3 site characteristic value listed in Table 2.4-208 (10% K_d) does not fall within (is less than) the SSAR site characteristic value. Measured K_d values are presented in Table 2.4-207 and show that the Unit 3 site characteristic value is conservative. See Section 2.4.13 for the radionuclide transport analysis.
	Sr-90	No value provided	**SSAR Table 1.9-1** 15 cm^3/g	The SSAR Table 1.9-1 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is consistent with SSAR Table 2.4-20.
NAPS ESP VAR 2.0-5e			**Unit 3** 2.08 cm^3/g	The Unit 3 site characteristic value listed in Table 2.4-208 (10% K_d) does not fall within (is less than) the SSAR site characteristic value. Measured K_d values are presented in Table 2.4-207 and show that the Unit 3 site characteristic value is conservative. See Section 2.4.13 for the radionuclide transport analysis.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-2	**Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value**			
	Distribution Coefficients (K_d) (continued)			
	Ru-106	No value provided	**SSAR Table 1.9-1** 55 cm^3/g	The SSAR Table 1.9-1 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is consistent with SSAR Table 2.4-20.
NAPS ESP VAR 2.0-5f			**Unit 3** 28.75 cm^3/g	The Unit 3 site characteristic value listed in Table 2.4-208 (10% K_d) does not fall within (is less than) the SSAR site characteristic value. Measured K_d values are presented in Table 2.4-207 and show that the Unit 3 site characteristic value is conservative. See Section 2.4.13 for the radionuclide transport analysis.
	Cs-134	No value provided	**SSAR Table 1.9-1** 30 cm^3/g	The SSAR Table 1.9-1 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is consistent with SSAR Table 2.4-20.
NAPS ESP VAR 2.0-5g			**Unit 3** 22.51 cm^3/g	The Unit 3 site characteristic value listed in Table 2.4-208 (10% K_d) does not fall within (is less than) the SSAR site characteristic value. Measured K_d values are presented in Table 2.4-207 and show that the Unit 3 site characteristic value is conservative. See Section 2.4.13 for the radionuclide transport analysis.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-2

Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
	Distribution Coefficients (K_d) (continued)			
	Cs-137	No value provided	**SSAR Table 1.9-1** 30 cm^3/g	The SSAR Table 1.9-1 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is consistent with SSAR Table 2.4-20.
NAPS ESP VAR 2.0-5h			**Unit 3** 22.51 cm^3/g	The Unit 3 site characteristic value listed in Table 2.4-208 (10% K_d) does not fall within (is less than) the SSAR site characteristic value. Measured K_d values are presented in Table 2.4-207 and show that the Unit 3 site characteristic value is conservative. See Section 2.4.13 for the radionuclide transport analysis.
	Y-90	No value provided	**SSAR Table 1.9-1** No value provided **Unit 3** 15.08 cm^3/g	SSAR Table 1.9-1 does not identify a distribution coefficient for this radionuclide. The Unit 3 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is listed in Table 2.4-208 (10% K_d). See Section 2.4.13 for the radionuclide transport analysis.
	Ni-63	No value provided	**SSAR Table 1.9-1** No value provided **Unit 3** 65.30 cm^3/g	SSAR Table 1.9-1 does not identify a distribution coefficient for this radionuclide. The Unit 3 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is listed in Table 2.4-208 (10% K_d). See Section 2.4.13 for the radionuclide transport analysis.
	Ag-110m	No value provided	**SSAR Table 1.9-1** No value provided **Unit 3** 14.71 cm^3/g	SSAR Table 1.9-1 does not identify a distribution coefficient for this radionuclide. The Unit 3 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is listed in Table 2.4-208 (10% K_d). See Section 2.4.13 for the radionuclide transport analysis.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-2

Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value			
Distribution Coefficients (K_d) (continued)			
Ce-144	No value provided	**SSAR Table 1.9-1** No value provided **Unit 3** 138.99 cm^3/g	SSAR Table 1.9-1 does not identify a distribution coefficient for this radionuclide. The Unit 3 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is listed in Table 2.4-208 (10% K_d). See Section 2.4.13 for the radionuclide transport analysis.
Np-239	No value provided	**SSAR Table 1.9-1** No value provided **Unit 3** 0.96 cm^3/g	SSAR Table 1.9-1 does not identify a distribution coefficient for this radionuclide. The Unit 3 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is listed in Table 2.4-208 (10% K_d). See Section 2.4.13 for the radionuclide transport analysis.
Pu-239	No value provided.	**SSAR Table 1.9-1** No value provided **Unit 3** 84.59 cm^3/g	SSAR Table 1.9-1 does not identify a distribution coefficient for this radionuclide. The Unit 3 site characteristic value is the distribution coefficient used to assess subsurface hydrological radionuclide transport and is listed in Table 2.4-208 (10% K_d). See Section 2.4.13 for the radionuclide transport analysis.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-2	**Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value**			
	Dose Consequences			
NAPS ESP VAR 2.0-6	Post Accident	No value provided	**SSAR Table 1.9-1** 10 CFR 50.34(a)(1) and 10 CFR 100 dose limits **Unit 3** 10 CFR 50.34(a)(1) and 10 CFR 100 dose limits	The Unit 3 site characteristic criteria fall within (are the same as) the SSAR Table 1.9-1 site characteristic criteria. SSAR Table 1.9-1 states that the radiological dose consequences due to gaseous releases from postulated plant accidents are addressed in SSAR Sections 15.2 and 15.4. SSAR Section 15.2 provides the site-specific χ/Q values for accident evaluations. The Unit 3 values are provided under Meteorological Dispersion (χ/Q) in Part 1 of this table above and the values fall within (are the same as) the SSAR Table 1.9-1 (SSAR Section 15.2) values. SSAR Section 15.4 provides dose estimates for three reactors. The estimates for the ABWR and AP-1000 do not apply to Unit 3. SSAR Section 15.4 provides estimated doses for postulated ESBWR design basis accidents (DBAs). Since the SSAR was submitted, activity releases were revised for the ESBWR DBAs. The Unit 3 dose from each DBA is provided in DCD Section 15.4, which conservatively assumes DCD χ/Q values rather than the Unit 3 site-specific χ/Q values. The DCD χ/Q values bound the Unit 3 values as shown under Meteorological Dispersion (χ/Q) in Part 1 of this table above. Most Unit 3 doses do not fall within (are larger than) the SSAR Table 1.9-1 (SSAR Section 15.4) values. While, the Unit 3 doses based on the DCD values are below the regulatory limits, this is NAPS ESP VAR 2.0-6.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-2

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value			
Release Point			
Minimum Distance to Site Boundary	No value provided	**SSAR Table 1.9-1** 870.17 m (2854.9 ft) **Unit 3** 870.17 m (2854.9 ft)	The Unit 3 site characteristic value falls within (is the same as) the SSAR Table 1.9-1 site characteristic value. SSAR Figure 2.1-1 identifies this distance as the closest point from the proposed facility boundary to the EAB. The facility boundary is the basis for estimating values for χ/Q values used in the SSAR and remains the basis for the Unit 3 site-specific χ/Q values. Figure 2.0-205 shows that Unit 3 power block buildings which could have postulated accidental fission product releases are located within that boundary. Because the buildings are within the boundary, the minimum distance to the site boundary is conservatively estimated. DCD Figure 2A-1 shows the potential release points for the Unit 3 power block buildings.
Population Density			
Population density at the time of initial site approval and within about 5 years thereafter	No value provided	**SSAR Table 1.9-1** Population density meets the guidance of RS-002, Section 2.1.3 for RG 4.7, Regulatory Position C.4 **Unit 3** Population density meets the guidance of RS-002, Section 2.1.3 for RG 4.7, Regulatory Position C.4	Based on SSAR Table 1.9-1, the Unit 3 site characteristic criterion is that at the time of initial site approval and within about 5 years hereafter, the population densities, including weighted transient population, averaged over any radial distance out to 20 miles (cumulative population at a distance divided by the circular area at that distance), would not exceed 500 persons per square mile. The Unit 3 site characteristic criterion falls within (is the same as) the SSAR Table 1.9-1 criterion. Time dependent population densities are provided in SSAR Section 2.1.3.6 which refers to SSAR Figure 2.1-14. That figure shows the projected population density at 2040 (i.e., much later than 5 years after expected initial site approval) meets the requirement.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-2

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value			
Population density at the time of initial operation	No value provided	**SSAR Table 1.9-1** Population density meets the guidance of RS-002, Section 2.1.3 **Unit 3** Population density meets the guidance of RS-002, Section 2.1.3	Based on SSAR Table 1.9-1, the Unit 3 site characteristic criterion is that the population densities, including weighted transient population, averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), would not exceed 500 persons per square mile at the time of initial operation. The Unit 3 site characteristic criterion falls within (is the same as) the SSAR Table 1.9-1 criterion. Time dependent population densities are provided in SSAR Section 2.1.3.6 which refers to SSAR Figure 2.1-14. That figure shows the projected population density at 2040 (i.e., much later than the expected time of initial operation) meets the requirement.
Population Density (continued)			
Population density over the lifetime of the new units until 2065	No value provided	**SSAR Table 1.9-1** Population density meets the guidance of RS-002, Section 2.1.3 **Unit 3** Population density meets the guidance of RS-002, Section 2.1.3	Based on SSAR Table 1.9-1, the Unit 3 site characteristic criterion is that the population densities, including weighted transient population, averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), would not exceed 1000 persons per square mile over the lifetime of Unit 3. The Unit 3 site characteristic criterion falls within (is the same as) the SSAR Table 1.9-1 criterion. Time dependent population densities are provided in SSAR Section 2.1.3.6 which refers to SSAR Figure 2.1-14. That figure shows the projected population density over the lifetime of Unit 3 operation meets the requirement.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-2

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value			
Site is Away from Very Densely Populated Centers	No value provided	**SSAR Table 1.9-1** 10 CFR 100.21(h) Meets requirement **Unit 3** 10 CFR 100.21(h) Meets requirement	Based on SSAR Table 1.9-1, the Unit 3 site characteristic criterion is that reactor sites should be located away from very densely populated centers. Areas of low population density are, generally, preferred. However, in determining the acceptability of a particular site located away from a very densely populated center but not in an area of low density, consideration will be given to safety, environmental, economic, or other factors, which may result in the site being found acceptable. The Unit 3 site characteristic criterion falls within (is the same as) the SSAR Table 1.9-1 criterion. SSAR Section 2.1.3.5 identifies that the nearest population center with more than 25,000 residents is the City of Charlottesville which is 36 miles away.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
NAPS SUP 2.0-2	**Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value**			
	Design Parameter	In the following rows, values for Unit 3 design characteristics presented in the DCD are identified in the Evaluation column	In the following rows, this column contains SSAR Table 1.9-1, Design Parameters and Unit 3 Design Characteristics	
	Structure Height	See Evaluation column	**SSAR Table 1.9-1** ≤71.3 m (234 ft) **Unit 3** 71.3 m (234 ft)	The tallest power block building is the turbine building (see DCD Figure 1.2-20) at 57.9 m (190 ft) above finished grade. The height of 57.9 m (190 ft) is based on the highest structural elevation of 60 m (196.85 ft) and a finished ground level grade of 4.5 m (14.76 ft), yielding a height of 55.5 m (182.09 ft), not including the parapet. The parapet of 1 m (3.28 ft) height is added to this for a total height above finished ground level grade of 56.5 m (185.37 ft). This value is rounded to 190 ft. The tallest power block structure is the Turbine Building vent stack (see DCD Table 2A-3) at 71.3 m (234 ft) above finished grade. This is the Unit 3 design characteristic value. The Unit 3 design characteristic value falls within (is equal to) the SSAR Table 1.9-1 design parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

NAPS SUP 2.0-2

Subject (17)	DCD Site Parameter Value (1)(17)	Site Characteristic	Evaluation
Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value			
Structure Foundation Embedment	See Evaluation column	**SSAR Table 1.9-1** ≤42.7 m (140 ft) **Unit 3** 20 m (65.6 ft) Nominal	The Unit 3 design characteristic value for structure foundation embedment is based on the bottom of the deepest power block structure basemat, which is the reactor building at 20 m (65.62 ft) nominal, below finished ground level grade (El. 88.24 m (289.50 ft)). The embedment of 20 m (65.62 ft) is based on the lowest elevation of -15.5 m (50.85 ft) and a finished ground level grade of +4.5 m (14.76 ft), yielding a depth of 20 m (65.62 ft), not including lean concrete below the basemat. This Unit 3 design characteristic value is shown in Table 2.5-213. The Unit 3 design characteristic value falls within (is less than) the SSAR Table 1.9-1 design parameter value.
Normal Plant Heat Sink Unit 3 Closed-Cycle, Dry and Wet Tower			
Make-Up Flow Rate	No value provided	**SSAR Table 1.9-1** ≤84.30 m^3/m (22,269 gpm) maximum (EC mode) **Unit 3** 84.26 m^3/m (22,260 gpm) maximum (EC mode)	The Unit 3 design characteristic value for the hybrid cooling tower makeup rate in EC mode is the expected rate of water withdrawal from Lake Anna to replace water lost from the operation of the tower during this mode. The losses are from evaporation, blowdown, and drift. The Unit 3 design characteristic value for the EC mode of operation falls within (is less than) the SSAR Table 1.9-1 design parameter value.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-2	**Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value**			
	Blowdown Flow Rate	No value provided	**SSAR Table 1.9-1** ≤21.1 m^3/m (5565 gpm) maximum (EC mode) **Unit 3** 21.0 m^3/m (5558 gpm) maximum (EC mode)	The Unit 3 design characteristic value for the hybrid cooling tower blowdown rate is the expected rate at which water is lost through blowdown flow from the cooling tower system to the WHTF. The Unit 3 design characteristic value for the EC mode of operation falls within (is less than) the SSAR Table 1.9-1 design parameter value.
	Unit 4 Dry Cooling Towers			
	Evaporation Rate	No value provided	**SSAR Table 1.9-1** None or negligible (on the order of 1 gpm, average) **Unit 3** Not applicable	This design parameter is not applicable because a Unit 4 is not included in this FSAR.
	Make-Up Flow Rate	No value provided	**SSAR Table 1.9-1** None or negligible (on the order of 1 gpm, average) **Unit 3** Not applicable	This design parameter is not applicable because a Unit 4 is not included in this FSAR.

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

	Subject [17]	DCD Site Parameter Value [1][17]	Site Characteristic	Evaluation
NAPS SUP 2.0-2	**Part 3 – Evaluation of SSAR Bounding Site Characteristics and Design Parameters For Which There is No Corresponding ESP or DCD Value**			
	Release Point			
	Elevation (Post Accident)	No value provided	**SSAR Table 1.9-1** Ground level **Unit 3** Ground level	The Unit 3 design characteristic value is an assumed ground level release point elevation for radiological consequences for accident releases. The Unit 3 design characteristic value falls within (is the same as) the SSAR Table 1.9-1 design parameter value.
	Plant Characteristics			
	Megawatts Thermal	See Evaluation column	**SSAR Table 1.9-1** ≤4500 MWt **Unit 3** 4500 MWt	This Unit 3 design characteristic value of 4500 MWt is the rated reactor thermal power, as described in DCD Section 1.1.2.7. The Unit 3 design characteristic value falls within (is the same as) the SSAR Table 1.9-1 design parameter value.

NAPS COL 2.0-1-A

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

1. The design of the Radwaste Building uses a set of design parameters that are specified in RG 1.143, Table 2, Class RW IIa instead of the corresponding values given in this table for all parameters except as follows: 1) Tornado: winds speeds, radius, pressure drop and rate of pressure drop; 2) Seismology: horizontal and vertical ground spectra: See DCD Figures 2.0-1 and 2.0-2.
2. Probable maximum flood level (PMF), as defined in Table 1.2-6 of Volume III of DCD Reference 2.0-4.
3. Maximum speed selected is based on Attachment I of DCD Reference 2.0-5, which summarizes the NRC Interim Position on RG 1.76. Concrete structures designed to resist Spectrum I missiles of SRP 3.5.1.4, Rev. 2, will also resist missiles postulated in RG 1.76, Revision 1.
4. Based on probable maximum precipitation (PMP) for one hour over 2.6 km^2 (one square mile) with a ratio of 5 minutes to one hour PMP of 0.32 as found in DCD Reference 2.0-3. Roof scuppers and drains are designed independently to limit water accumulation on the roof to no more than 100 mm (4 in) during PMP conditions. See also DCD Table 3G.1-2.
5. Maximum design roof load accommodates snow load and 48-hour probable maximum winter precipitation (PMWP) in DCD References 2.0-2 and 2.0-6. Roof scuppers and drains are designed independently to limit water accumulation on the roof to no more than an average depth of 100 mm (4 in) during PMWP conditions. See also DCD Table 3G.1-2.
6. ESBWR site parameter zero percent exceedance values are based on conservative estimates of historical high and low values for potential sites. Consistent with DCD Reference 2.0-4, they represent historical limits excluding peaks of less than two hours. One and two percent annual exceedance values were selected in order to bound the values presented in DCD Reference 2.0-4 and available Early Site Permit applications.
7. At foundation level of Seismic Category I structures. For minimum dynamic bearing capacity site-specific application, use the larger value or a linearly interpolated value of the applicable range of shear wave velocities at the foundation level. The shear wave velocities of soft, medium and hard soils are 300 m/sec (1000 ft/sec), 800 m/sec (2600 ft/sec) and greater than or equal to 1700 m/sec (5600 ft/sec), respectively.

NAPS COL 2.0-1-A

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

8. This is the equivalent uniform shear wave velocity (V_{eq}) over the entire soil column at seismic strain, which is a lower bound value after taking into account uncertainties. V_{eq} is calculated to achieve the same wave traveling time over the depth equal to the embedment depth plus 2 times the largest foundation plan dimension below the foundation as follows:

$$V_{eq} = \frac{\sum d_i}{\sum \frac{d_i}{V_i}}$$

where d_i and V_i are the depth and shear wave velocity, respectively, of the i^{th} layer. The ratio of the largest to the smallest shear wave velocity over the mat foundation width at the foundation level does not exceed 1.7.

9. Safe Shutdown Earthquake (SSE) design ground response spectra of 5% damping, also termed Certified Seismic Design Response Spectra (CSDRS), are defined as free-field outcrop spectra at the foundation level (bottom of the base slab) of the Reactor/Fuel and Control Building structures. For ground surface founded Firewater Service Complex structures, the CSDRS is 1.35 times the values shown in DCD Figures 2.0-1 and 2.0-2.

10. Values reported here are actually design criteria rather than site parameters. They are included here because they don't appear elsewhere in the DCD.

11. If a selected site has a χ/Q value that exceeds the ESBWR reference site value, the COL applicant will address how the radiological consequences associated with the controlling design basis accident continue to meet the dose reference values provided in 10 CFR 50.34(a) and control room operator dose limits provided in General Design Criterion 19 using site-specific χ/Q values.

12. If a selected site has χ/Q values that exceed the ESBWR reference site values, the release concentrations in DCD Table 12.2-17 would be adjusted proportionate to the change in χ/Q values using the stack release information in DCD Table 12.2-16. In addition, for a site selected that exceeds the bounding χ/Q or D/Q values, the COL applicant will address how the resulting annual average doses (DCD Table 12.2-18b) continue to meet the dose reference values provided in 10 CFR 50 Appendix I using site-specific χ/Q and D/Q values.

13. Value was selected to comply with expected requirements of southeastern coastal locations.

14. Localized liquefaction potential under other than Seismic Category I structures is addressed per SRP 2.5.4 in DCD Table 2.0-2.

NAPS COL 2.0-1-A

Table 2.0-201 Evaluation of Site/Design Parameters and Characteristics

15. Settlement values are long-term (post-construction) values except for differential settlement within the foundation mat. The design of the foundation mat accommodates immediate and long-term (post-construction) differential settlements after the installation of the basemat.
16. For sites not meeting the soil property requirements, a site-specific analysis is required.
17. Information in this column and notes (1) through (16) are from DCD Table 2.0-1. In these notes, “DCD” was added before cited tables, figures, and references from the DCD.

NAPS COL 2.0-1-A

Table 2.0-202 Comparison of ESP and Unit 3 Horizontal Spectra for Zone III-IV

Freq. (Hz)	Unit 3 SA (g)(1)	ESP SA (g)(2)	Controlling ESP or Unit 3	% Difference
100	0.448	0.555	ESP Spectra	-19.3
50	0.969	1.195	ESP Spectra	-18.9
30	1.206	1.47	ESP Spectra	-18.0
25	1.193	1.476	ESP Spectra	-19.2
20	1.163	1.446	ESP Spectra	-19.6
10	0.877	0.945	ESP Spectra	-7.20
8	0.687	0.717	ESP Spectra	-4.18
6	0.468	0.481	ESP Spectra	-2.70
5	0.367	0.376	ESP Spectra	-2.39
4	0.283	0.287	ESP Spectra	-1.39
3	0.214	0.214	ESP Spectra	0.00
2.5	0.18	0.179	Unit 3 Spectra	0.56
2	0.143	0.142	Unit 3 Spectra	0.70
1	0.0676	0.0677	ESP Spectra	-0.15
0.8	0.0578	0.0576	Unit 3 Spectra	0.35
0.6	0.0492	0.0488	Unit 3 Spectra	0.82
0.5	0.0432	0.0429	Unit 3 Spectra	0.70
0.4	0.0344	0.0343	Unit 3 Spectra	0.29
0.3	0.0234	0.0233	Unit 3 Spectra	0.43
0.2	0.0131	0.01298	Unit 3 Spectra	0.92
0.1	0.00386	0.00382	Unit 3 Spectra	1.05

(1) Values from Table 2.5-201

(2) Values from SSAR Table 2.5-27A

NAPS COL 2.0-1-A

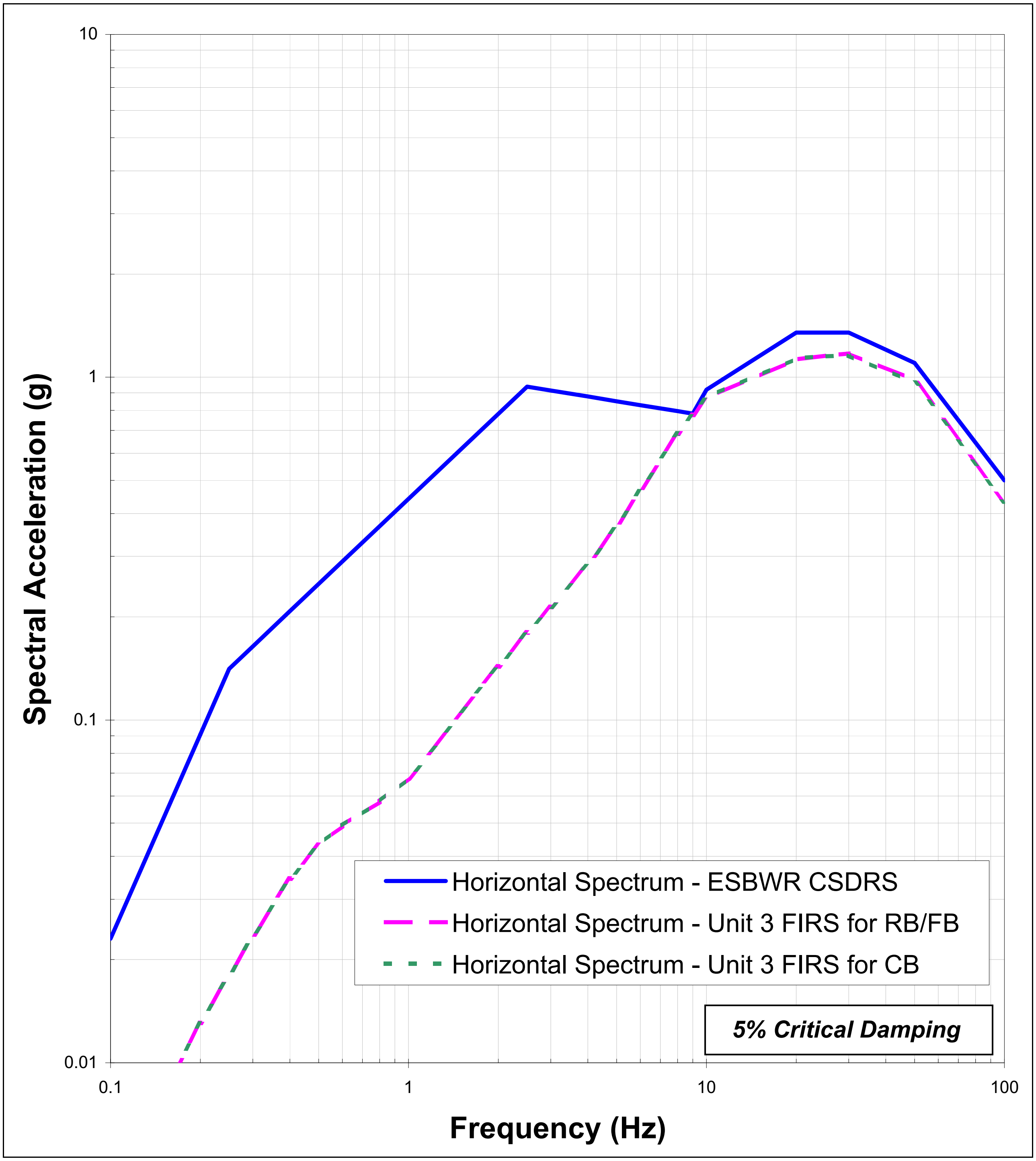
Table 2.0-203 Comparison of ESP and Unit 3 Vertical Spectra for Zone III-IV

Freq. (Hz)	Unit 3 SA (g)(1)	ESP SA (g)(2)	Controlling ESP or Unit 3	% Difference
100	0.448	0.555	ESP Spectra	-19.3
50	1.085	1.33	ESP Spectra	-18.4
30	1.134	1.38	ESP Spectra	-17.8
25	1.050	1.29	ESP Spectra	-18.6
20	0.965	1.2	ESP Spectra	-19.6
10	0.658	0.708	ESP Spectra	-7.06
8	0.515	0.537	ESP Spectra	-4.10
6	0.351	0.36	ESP Spectra	-2.50
5	0.275	0.282	ESP Spectra	-2.48
4	0.212	0.215	ESP Spectra	-1.40
3	0.161	0.16	Unit 3 Spectra	0.63
2.5	0.135	0.134	Unit 3 Spectra	0.75
2	0.107	0.106	Unit 3 Spectra	0.94
1	0.0507	0.0507	ESP Spectra	0.00
0.8	0.0434	0.0432	Unit 3 Spectra	0.46
0.6	0.0369	0.0366	Unit 3 Spectra	0.82
0.5	0.0324	0.0321	Unit 3 Spectra	0.93
0.4	0.0258	0.0257	Unit 3 Spectra	0.39
0.3	0.0176	0.0174	Unit 3 Spectra	1.15
0.2	0.00983	0.00973	Unit 3 Spectra	1.03
0.1	0.00290	0.00286	Unit 3 Spectra	1.40

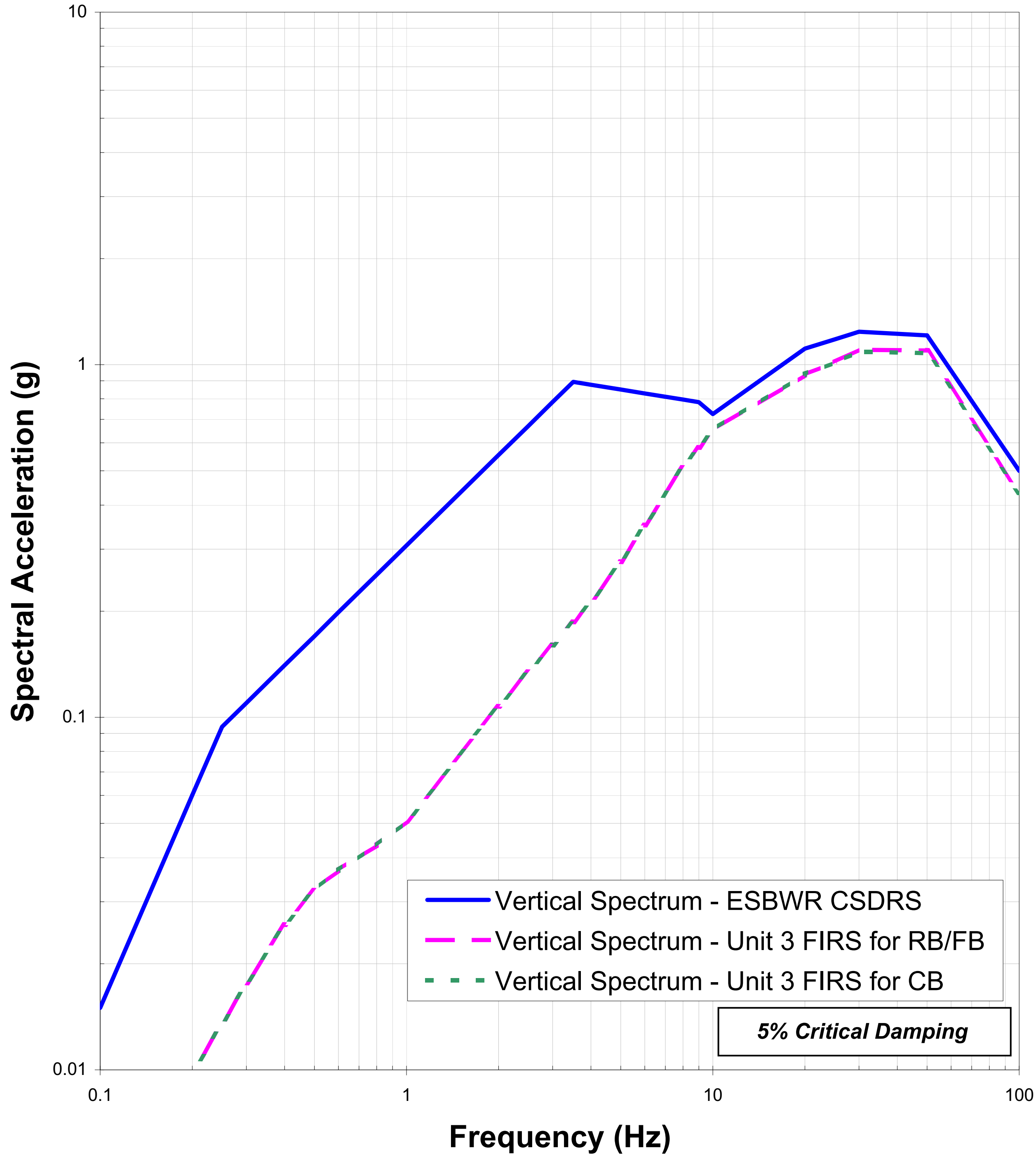
(1) Values from Table 2.5-201

(2) Values from SSAR Table 2.5-27A

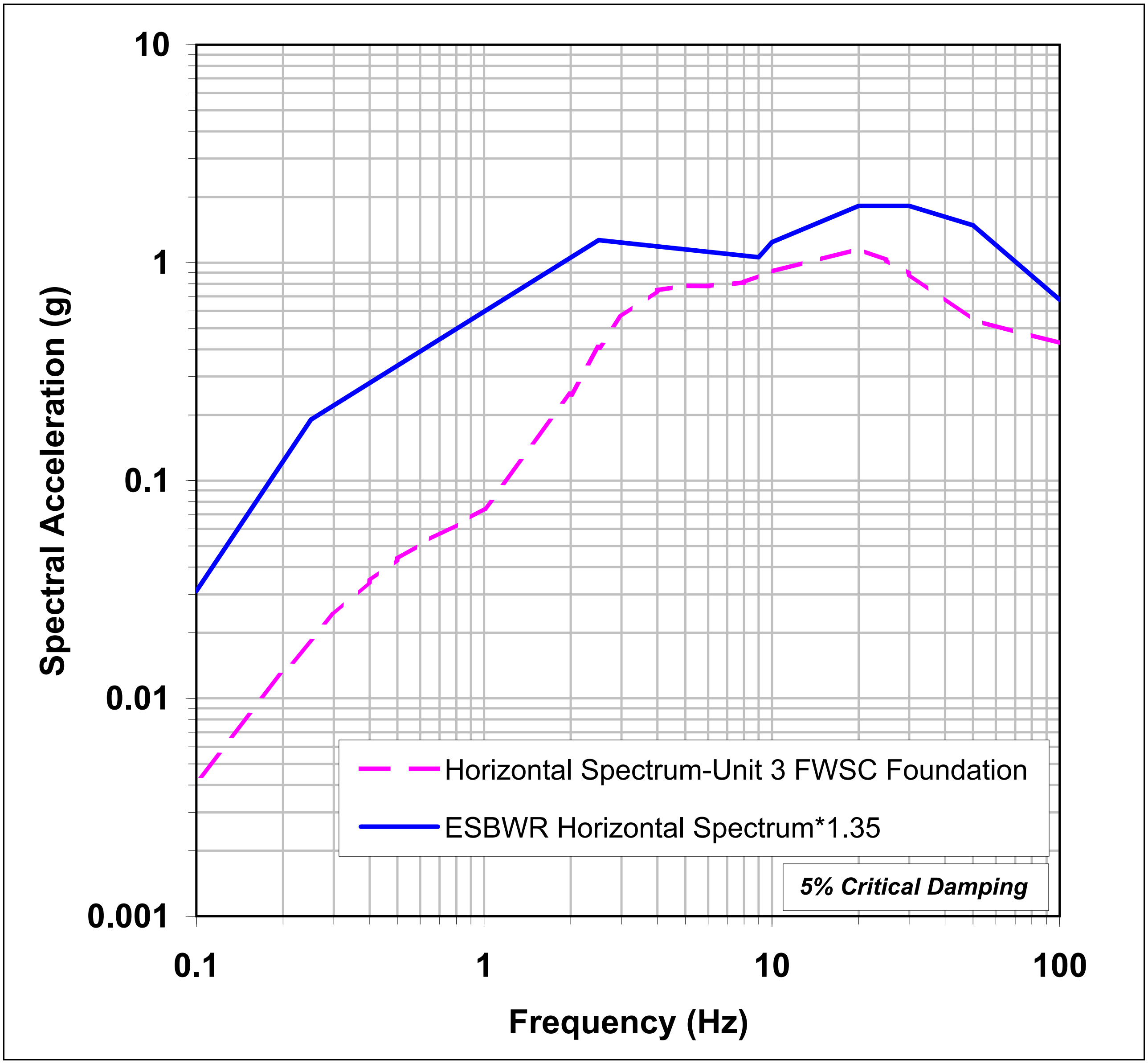
NAPS COL 2.0-1-A **Figure 2.0-201 Comparison of Horizontal CSDRS with Unit 3 FIRS for the Reactor Building/Fuel Building and Control Building**



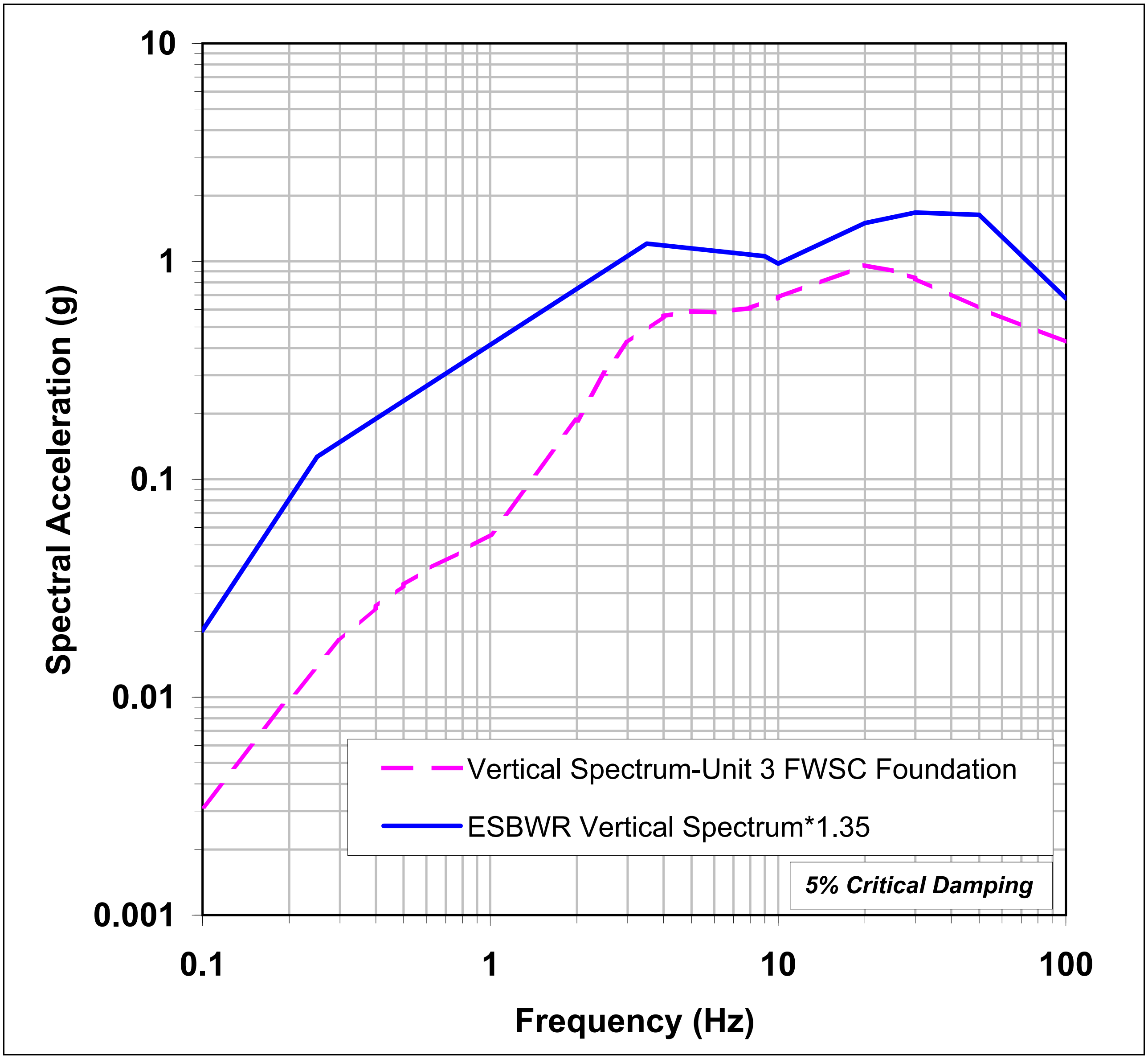
NAPS COL 2.0-1-A **Figure 2.0-202 Comparison of Vertical CSDRS with Unit 3 FIRS for the Reactor Building/Fuel Building and Control Building**



NAPS COL 2.0-1-A **Figure 2.0-203 Comparison of Horizontal CSDRS with Unit 3 FIRS for the FWSC**

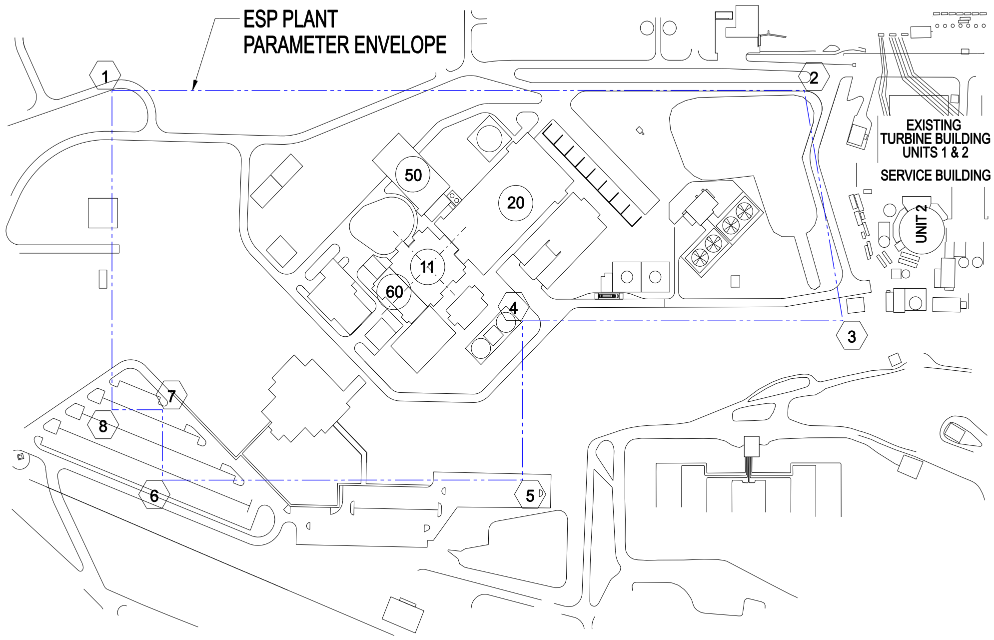


NAPS COL 2.0-1-A **Figure 2.0-204 Comparison of Vertical CSDRS with the Unit 3 FIRS for the FWSC**



NAPS COL 2.0-1-A

Figure 2.0-205 Unit 3 Power Block Building Locations Within the ESP Proposed Facility Boundary



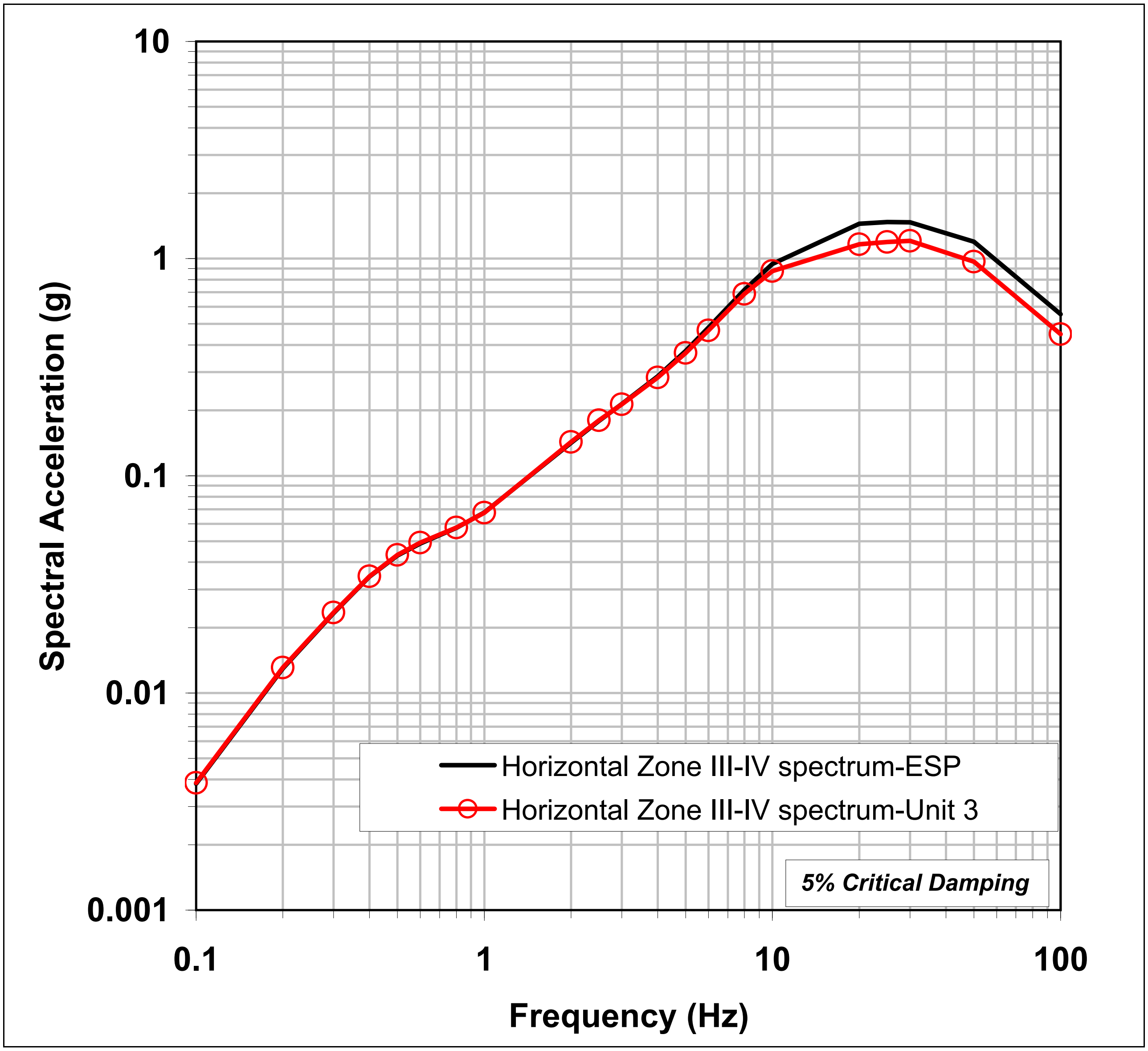
LEGEND

11 REACTOR BUILDING
20 TURBINE BUILDING
50 RADWASTE BUILDING
60 FUEL BUILDING

	COORDINATES (NAPS U1 & U2 GRID)		COORDINATES (STATE PLANE NAD 83 VA SOUTH ZONE)	
POINT	PLANT NORTH (FEET)	PLANT EAST (FEET)	NORTHING (FEET)	EASTING (FEET)
1	5400.00	7320.00	3909903.70	11684870.43
2	5400.00	9378.00	3910657.08	11686785.57
3	4741.66	9492.26	3910086.27	11687132.90
4	4741.66	8539.65	3909737.54	11686246.42
5	4287.00	8539.65	3909314.44	11686412.86
6	4287.00	7470.00	3908922.87	11685417.46
7	4488.00	7470.00	3909109.92	11685343.88
8	4488.00	7320.00	3909055.00	11685204.29

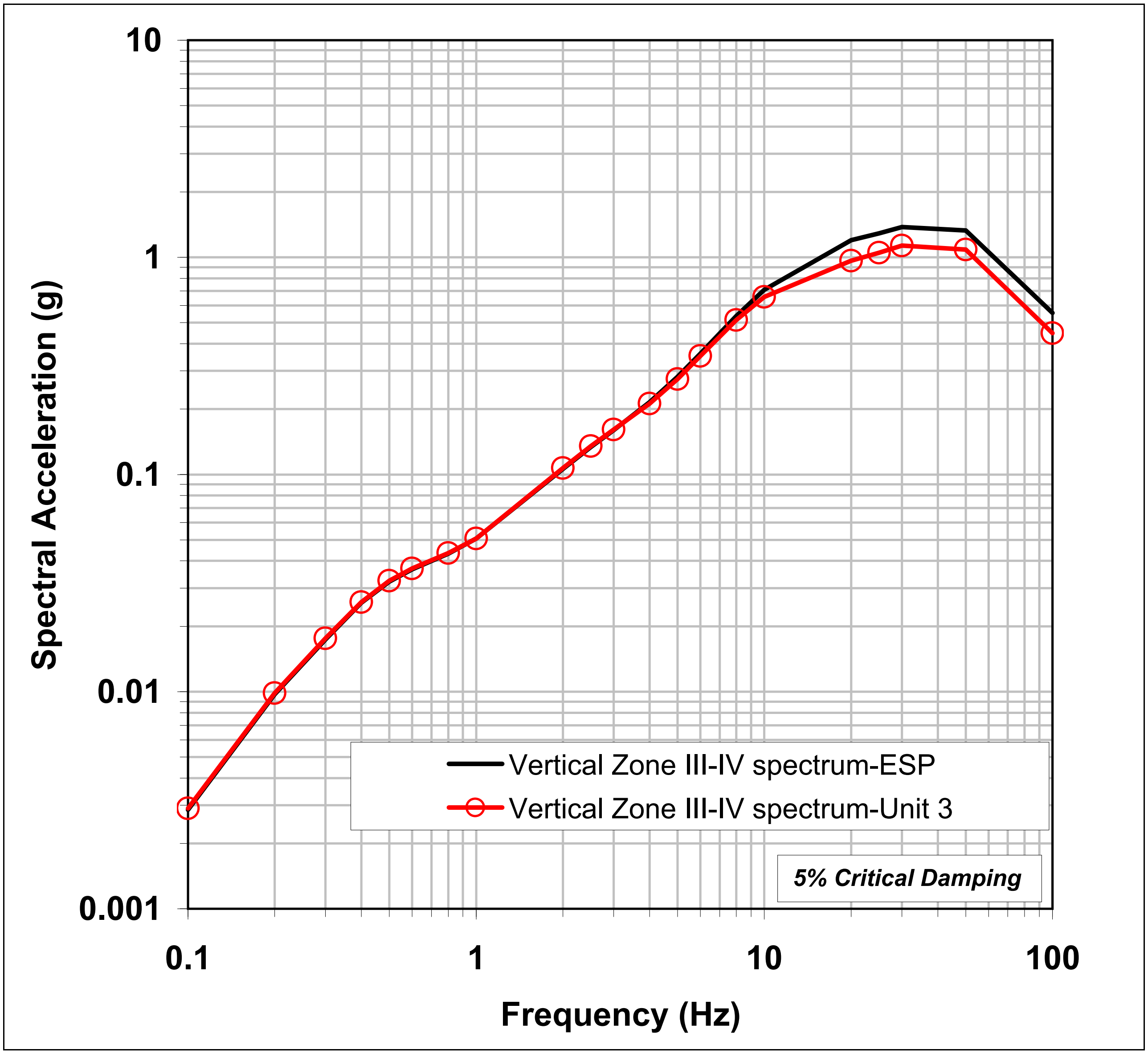
NAPS COL 2.0-1-A

Figure 2.0-206 Comparison of ESP and Unit 3 Horizontal SSE Design Response Spectra



NAPS COL 2.0-1-A

Figure 2.0-207 Comparison of ESP and Unit 3 Vertical SSE Design Response Spectra



2.1 Introduction

2.1.1 Site Location and Description

NAPS COL 2.0-2-A

The information needed to address DCD COL Item 2.0-2-A is included in SSAR Section 2.1.1, which is incorporated by reference with the following supplements.

2.1.1.1 Site Location

The first paragraph of this SSAR section is supplemented as follows with information on the location of Unit 3 at the NAPS site.

NAPS COL 2.0-2-A

The Unit 3 site plan is shown in Figure 2.1-201 and remains within the ESP proposed facility boundary (ESP plant parameter envelope) as shown in Figure 2.0-205. The center of the Unit 3 Reactor Building is approximately 450 m (1476 ft) southwest of the center of the Unit 2 Containment Building.

NAPS ESP COL 2.1-1

The coordinates of the Unit 3 Reactor Building are:

- Latitude 38 Degrees 03 Minutes 31.01 Seconds (38.058614)
- Longitude 77 Degrees 47 Minutes 41.80 Seconds (77.794944)

The corresponding Universal Transverse Mercator (UTM) coordinates are:

- NAD83, Zone 18-78W to 72W (US ft), N13832016.995/E835901.295

2.1.1.2 Site Description

The last paragraph of this SSAR section is supplemented as follows with information on ownership and control.

NAPS COL 2.0-2-A

Since the ESP Application was submitted by Dominion Nuclear North Anna, LLC, the Commonwealth of Virginia has passed legislation re-regulating the electric power industry in Virginia, and the State Corporation Commission has determined that Dominion should be the COL applicant. In addition, ODEC has elected to participate in the ownership of Unit 3. As a result, rather than Dominion Nuclear North Anna, LLC, purchasing or leasing the ESP Site, Dominion and ODEC continue to jointly own the entire NAPS station, including Unit 3, and Dominion continues to control the existing exclusion area as a single exclusion area and single restricted area for all reactor units located within the NAPS property, including Unit 3.

2.1.2 Exclusion Area Authority and Control

NAPS COL 2.0-3-A The information needed to address DCD COL Item 2.0-3-A is included in SSAR Section 2.1.2, which is incorporated by reference with the following supplements.

2.1.2.1 Authority

The first four paragraphs in this SSAR section are supplemented as follows with information to address the authority of the COL applicant.

NAPS COL 2.0-3-A Since the ESP Application was submitted by Dominion Nuclear North Anna, LLC, the Commonwealth of Virginia has passed legislation re-regulating the electric power industry in Virginia, and the State Corporation Commission has determined that Dominion should be the COL applicant. In addition, ODEC has elected to participate in the ownership of Unit 3. As a result, rather than Dominion Nuclear North Anna, LLC, purchasing or leasing the ESP Site, Dominion and ODEC will continue to jointly own the entire NAPS site, including Unit 3, and Dominion will continue to maintain sole control of the existing exclusion area as a single exclusion area and single restricted area for the all reactor units located within the NAPS property, including Unit 3.

NAPS ESP PC 3.E(1) Dominion currently controls the NAPS site and exclusion area under its existing agreement with ODEC, and no approvals are required by state law for shared control of the exclusion area.

As the owners of NAPS, Dominion and ODEC possess the right to implement the site redress plan.

The last paragraph in this SSAR section is supplemented as follows with information to address recreational use of the lake.

NAPS COL 2.0-3-A The lake access and control practices in effect for Units 1 and 2 are maintained for Unit 3.

2.1.2.2 Control of Activities Unrelated to Plant Operation

The third paragraph in this SSAR section is supplemented as follows with information to address arrangements with appropriate agencies for emergencies.

NAPS ESP COL 2.1-2

Under the Commonwealth of Virginia's Radiological Emergency Response Plan (COVRERP) (Reference 2.1-201), the Virginia Department of Game and Inland Fisheries is responsible for warning people in boats and assisting in traffic control of boats on Lake Anna in the vicinity of NAPS. This arrangement is documented in the COVRERP, Appendix 1.

2.1.3 Population Distribution

NAPS COL 2.0-4-A

The information needed to address DCD COL Item 2.0-4-A is included in SSAR Section 2.1.3, which is incorporated by reference.

Section 2.1 References

2.1-201 Commonwealth of Virginia's Radiological Emergency Response Plan (COVRERP), May 2007.

NAPS COL 2.0-2-A

Figure 2.1-201 Site Plan with Topography

NOTES:

1. DESIGN PLANT GRADE ELEVATION FOR POWER BLOCK AREA IS 88.4m (290 FT) (MSL)

2.2 Nearby Industrial, Transportation, and Military Facilities

NAPS COL 2.0-5-A

The information needed to address DCD COL Item 2.0-5-A is included in SSAR Sections 2.2.1 and 2.2.2, which are incorporated by reference with the following supplements. SSAR Section 3.5.1.6 is also incorporated by reference, with no supplements.

2.2.2.1 Industrial Facilities

The first paragraph of this SSAR section is supplemented as follows with information on nearby industrial facilities.

NAPS ESP COL 2.2-1

Since the SSAR was submitted, no hazardous industrial facilities have been added at the 2.51 km^2 (620 acres) industrial development near the Unit 3 EAB. The industrial site poses no hazard to Unit 3.

2.2.2.6.1 Airports

The first paragraph of this SSAR section is supplemented as follows with information to identify an additional airport in the vicinity of Unit 3.

NAPS COL 2.0-5-A

A third airport within 16.1 km (10 mi) of the Unit 3 site opened in 2007. Table 2.2-201 provides operations-related information. The location is shown with other nearby airports in Figure 2.2-201. Because this is a small private airport, it is not expected to grow substantially in the foreseeable future.

After the fourth paragraph of this SSAR section, a new paragraph is added to describe the additional airport in the vicinity of Unit 3.

Seven Gables, a private landing strip with an unlighted 457 m (1500 ft) turf runway, is approximately 12.2 km (7.6 mi) north-northwest of the site. It is not licensed for commercial use and with only three small aircraft based on the field (one single-engine airplane, one helicopter, and one ultralight), the expected volume of traffic is very light. (Reference 2.2-201)

2.2.2.6.2 Airways

The first paragraph of this SSAR section is supplemented as follows with information to identify an additional military training flight in the vicinity of NAPS.

NAPS COL 2.0-5-A

One civil airway (V223) and four military training routes (IR714, IR760, VR1754, and VR1755) pass near the Unit 3 site as shown in Figure 2.2-201, which is based on the Washington Sectional Aeronautical Chart issued in 2007 (Reference 2.2-202). The U.S. Department of the Navy identifies a total of 341 flight operations in the year 2006 for the four routes (Reference 2.2-203), as compared to the SSAR assumption of 6000 flights per year. As a result, the number of military training flights assumed in the SSAR remains bounding.

The second paragraph of this SSAR section is supplemented as follows with information on distances from military training flight routes to Unit 3.

The centerlines of three of the military training routes IR714, IR760, and VR1754, which are 16.1 km (10 mi) across, lie within 1.6 km (1 mi) of the Unit 3 site. The centerline of the fourth military training route, VR1755, is more than 12.9 km (8 mi) from Unit 3.

2.2.3 Evaluation of Potential Accidents

NAPS COL 2.0-6-A

The information needed to address DCD COL Item 2.0-6-A is included in SSAR Section 2.2.3, which is incorporated by reference with the following supplements.

2.2.3.1.1 Truck Traffic

Add the following at the end of this section.

NAPS COL 2.0-8-A

Gasoline tanker truck explosion hazards due to local deliveries on-site are addressed by considering the likelihood of an accident leading to a significant overpressure. According to RG 1.91, the risk from potential explosion hazards can be shown to be sufficiently low on the basis of low probability of an explosion when the rate of exposure to a peak overpressure in excess of 7 kPa (1 psi) is less than 10^{-6} per year using conservative assumptions. Per RG 1.91, the following equation was used:

$$r = n_1 \times n_2 \times f \times s \qquad (2.2.3.1.1\text{-}1)$$

where,

- r = exposure rate (the probability of an explosion occurring)
- n_1 = accidents per km (mi) for the transportation mode (truck transport)
- n_2 = cargo explosion per accident for the transportation mode
- f = frequency of shipment for the substance, in shipments per year
- s = exposure distance in km (mi)

The number of accidents per km (mi) for truck transport, n_1, is 1.25×10^{-6}/km (2×10^{-6}/mi) based on an average value for large trucks (References 2.2-213 and 2.2-214). This is comparable to the 2006 accident rate per mile for all vehicle types for the Commonwealth of Virginia. The national average accident rate includes accidents at highway speeds and those involving multiple vehicles. Whereas, under the controlled conditions on the NAPS site; specifically, supervised truck movements and low speed limits, the accident rate per mile would be much lower. Therefore, the use of 1.25×10^{-6}/km (2×10^{-6}/mi) as an estimate of the accident rate for tractor-trailers carrying hazardous materials is very conservative.

The probability of a release and cargo explosion per accident, n_2, is determined using the assumption that 20 percent of highway truck crashes result in releases/spills, 20 percent of those releases involve a complete release of total cargo (Reference 2.2-213), and the probability of ignition given a release is 1. This results in an overall number of cargo explosions per accident of 0.04 or 4 percent.

The frequency of shipment, f, for on-site delivery of gasoline to the North Anna site is two to three times per year. Conservatively assuming that there are two deliveries per unit per year, the addition of a third unit would increase the number of gasoline deliveries per year to six. Therefore, a value of six deliveries per year is used to determine the accident rate for onsite gasoline delivery by truck.

Considering the portions of on-site delivery truck routes within 580 m (1900 ft) of Unit 3 safety-related structures, the exposure distance, s, would be 2.61 km (1.62 mi). However, using 580 m (1900 ft) is conservative in comparison with the methodology described in Section 2.2.3.1.3 for determining the safe separation distance. Therefore, the exposure distance of 2.61 km (1.62 mi) is also conservative.

Using the conservative inputs to Equation 2.2.3.1.1-1 as described above, an annual exposure rate of 7.8×10^{-7} was obtained, which is less than 10^{-6} per year, so there is a sufficiently low risk from explosion during on-site gasoline tanker truck deliveries.

NAPS ESP COL 2.2-2

2.2.3.1.3 On-Site Chemicals

The chemical materials stored on-site at Units 1, 2, and 3 are identified in Table 2.2-202. This table also identifies storage locations and the quantity of each chemical/material. Properties relative to the hazards of each chemical and the results of a screening analysis based on these hazardous properties are provided in Table 2.2-203. The on-site chemicals with the potential to be flammable or explosive are evaluated for possible effects on Unit 3 safety-related SSCs.

Table 2.2-203 shows that the majority of the chemicals are not toxic. For chemicals with immediately dangerous to life or health (IDLH) values listed in this table, the effects of toxic vapors or gases and their potential for incapacitating Unit 3 control room operators are evaluated and the results presented in Section 6.4.

Table 2.2-203 also shows that very few chemicals present a flammability or explosive hazard. As shown by the table column labeled “Flammable/Explosive?”, three of the materials have flammability and explosive properties that needed analysis. These are hydrogen, hydrazine, and Nalco H-130© (a non-oxidizing biocide). The analysis of these materials is described below.

For each of these materials, minimum safe separation distances for flammable materials and explosive materials were determined for comparison with the actual distance from the storage location to the nearest Unit 3 safety-related SSC. For flammable materials, there are two minimum safe separation distances based on whether the material vaporizes and burns (thermal exposure hazard) or whether the material vaporizes and detonates (explosion overpressure hazard).

The safe separation distance for the storage of explosive materials is determined according to RG 1.91 and FM Global Guidelines for Evaluating the Effects of Vapor Cloud Explosions Using a TNT Equivalency Method (Reference 2.2-204).

Per RG 1.91, 7 kPa (1 psi) is a conservative value of peak positive incident overpressure, below which no significant damage to safety-related SSCs would be expected. The minimum safe separation distance for an explosive hazard is the distance from the location of storage to the point where an explosion results in less than a 7 kPa (1 psi) peak incident pressure. In determining this distance for each material, the following conservative assumptions were also used. A volume of vapor equal to the empty volume of the largest storage vessel was considered available for combustion and an explosion yield factor of 100 percent was used to address the possibility of an in-vessel confined explosion. This is conservative because only that small portion of the vapor within the flammability limits would be available for combustion and potential explosion.

The two minimum safe separation distances for a flammable material (thermal exposure hazard and/or explosion overpressure hazard) were determined based on the following model. Flammable materials in the liquid state can evaporate and form an unconfined vapor cloud. Such a vapor cloud is assumed to drift towards Unit 3 before ignition occurs. Because a vapor cloud disperses as it travels downwind, there may be parts of a cloud where the vapor concentration is in the flammable range. These portions of a vapor cloud, between the lower flammability limit (LFL) and upper flammability limit (UFL), are assumed to burn when the cloud reaches an ignition source. The speed of the flame front through the vapor cloud determines whether the event is a deflagration or a detonation.

When a deflagration occurs, the hazard is from thermal exposure of the nearby surfaces from heat generated by the fire. A deflagration is assumed to be possible up to the point where the vapor cloud reaches the lower flammability limit of the material. The minimum safe separation distance for flammability hazard (thermal exposure) is the maximum distance from the storage site (the spill site) to the location where the vapor cloud can exist and still be between the UFL and the LFL.

Because a detonation would generate an explosive force, the possibility of a detonation is evaluated for each flammable material. The RG 1.91 limit of 7 kPa (1 psi) is again used as a conservative value of peak positive incident overpressure, below which no significant damage to safety-related SSCs would be expected. The minimum safe separation distance for a flammability hazard (explosion overpressure) is the distance from the storage site (the spill site) to the location where the assumed detonation of the traveled vapor cloud results in a peak incident pressure of no more than 7 kPa (1 psi).

In determining these distances for each material, the following model and conservative assumptions were also used. The on-site chemicals in Table 2.2-202 with an identified flammability range were modeled using the Areal Locations of Hazardous Atmospheres (ALOHA) air dispersion model (Reference 2.2-205). ALOHA determined the distances where the vapor cloud may exist between the LFL and the UFL, presenting the possibility of ignition, detonation, and potential overpressure effects. Conservative assumptions were used in the analyses. The meteorological assumptions were: F (stable) stability class with a wind speed of 1 m/sec (3.3 ft/sec); ambient temperature of 25°C (77°F); relative humidity, 50 percent; cloud cover, 50 percent; and atmospheric pressure, 1 atmosphere (14.7 psi). For each chemical analyzed, the model conservatively assumed that the maximum volume of the storage vessel leaked to form a 1 cm (0.4 inch) thick puddle. This provides a significant surface area to maximize evaporation and the formation of a vapor cloud.

Table 2.2-204 provides the safe separation distances for flammable and explosive materials and compares them to the actual distance to the nearest safety-related Unit 3 SSC. The results indicate that a fire or explosion from the identified hazardous chemicals and materials would not adversely affect the safe operation or shutdown of Unit 3.

2.2.3.2.2 **Airways**

The second and subsequent paragraphs of this SSAR section are supplemented as follows with information on effective plant areas for Unit 3 and the evaluation results.

NAPS COL 2.0-6-A

For the SSAR, which used a PPE approach, the type of reactor with the tallest reactor building height (71.323 m (234 ft) above grade) was evaluated. For Unit 3, the ESBWR Reactor Building, Control Building, Fuel Building, and Radwaste Building are evaluated. See DCD Figures 1.2-1 through 1.2-11 for the nuclear island (Reactor, Control, and Fuel Buildings) and DCD Figures 1.2-21 through 1.2-25 for the Radwaste Building. For flights in the civilian airway, a total effective plant area of 0.062 square kilometers (0.024 square miles) was used in the evaluation. For flights in the military airways, a total effective plant area of 0.083 square kilometers (0.032 square miles) was used in the evaluation.

For civil airway V223, the Unit 3 result is:

$$PFA = 6.37 \times 10^{-8}$$

For military routes, IR714, IR760, VR1754 and VR1755, the Unit 3 result is:

$$PFA = 3.84 \times 10^{-8}$$

The total of these two accident probabilities meets the NUREG-0800, Section 2.2.3 guideline and is of an order of magnitude of 10^{-7} per year.

NAPS COL 2.0-6-A

2.2.3.4 Fires

An accident in the vicinity of Unit 3 could lead to a fire, but the absence of industrial facilities, pipelines, and commercial navigation in the Unit 3 vicinity results in a low probability of chemical explosions and fires. Similarly, land transportation routes are some distance from the Unit 3 site and are unlikely to start a fire that affects Unit 3. The potential for off-site wildfires exists due to the rural nature of the NAPS site and presence of off-site vegetation to the west and south of the site.

The analysis of a wildfire near Unit 3 was performed using the methodology in NUREG-1805 (Reference 2.2-206) to determine the incident heat flux on Unit 3. The conservative assumptions in the analysis included the following:

- The wildfire is assumed to occur at plant elevation.
- The closest forest area with a significant fire line is southeast of the Unit 3 control building. The fire line is modeled as 134 m (440 ft) wide at a distance of 387 m (1270 ft) from the nearest safety-related structure, the Unit 3 Control Building.
- The wildfire burns through the forest toward Unit 3 in a uniform fire line perpendicular to the line of closest separation between the 134 m (440 ft) wide fire line and the Unit 3 Control Building. While more of the forested area could burn toward the south, using a wider fire line would increase the separation distance from the Unit 3 safety-related structures. The forest area that is burning is assumed to continuously and simultaneously burn at peak output.

The maximum incident heat flux from a wildfire at the Unit 3 Control Building is 0.5 kW/m^2. For comparison, this level of thermal radiation is about one third that of incident radiation from the sun on the earth, which is approximately 1.4 kW/m^2. Given the conservatism in the assumptions and the large separation distances to safety-related structures, a wildfire originating offsite would not affect the safe operation or shutdown of Unit 3.

In addition to a potential fire in the vicinity of Unit 3, a fire involving chemicals stored on the NAPS site was considered. Table 2.2-203 lists the chemicals and shows those which are potentially flammable or explosive. The stored hydrazine, liquid hydrogen, and Nalco H-130© non-oxidizing biocide were evaluated as potential fire hazards using ALOHA. The ALOHA analyses show that these materials are sufficiently separated from safety-related SSCs that further analysis is not required. Table 2.2-203 and the ALOHA results in Table 2.2-204 demonstrate that significant effects are not expected due to a fire involving onsite chemicals and fuels.

2.2.3.5 Collisions with the Unit 3 Intake Structure

The Unit 3 intake structure is located on Lake Anna in a cove behind a cofferdam that is northeast of the Unit 3 power block area as shown in Figure 2.1-201. Lake Anna has small pleasure boats used solely for recreation. There are no large boats or barges on the lake. The area around the Unit 3 intake structure is managed by Dominion as a part of the exclusion area. Due to the presence of the cofferdam, there is no potential for a collision between a boat on Lake Anna and the Unit 3 intake structure. Also, because the Unit 3 intake structure is not a safety-related structure, such a collision could not affect the safe operation or shutdown of Unit 3.

2.2.3.6 Liquid Spills Near the Intake Structure

An accidental spill of an oil or liquid in Lake Anna near the Unit 3 intake structure that may be corrosive, cryogenic, or a coagulant was considered and determined to not be credible or have a low probability of occurrence and have no consequences for the safety of Unit 3. Lake Anna has small pleasure boats for recreational use. There are no large boats or barges. The only liquids with the potential to be spilled are motor oil and gasoline fuel from a small pleasure boat. The quantities in such spills would be very small. The oil or gasoline from a spill would float on the Lake Anna surface while the openings in the Unit 3 intake channel culverts through the cofferdam are underwater. Therefore, such spills could not affect the safe operation or shutdown of Unit 3.

2.2.3.7 Effects of Design Basis Events

As concluded in the previous sections, no events are identified that are likely to occur and have potential consequences that affect the safety of Unit 3. The potential consequences associated with the on-site hazards of stored chemicals are not significant. None of the scenarios are serious enough to affect the safety of Unit 3 to the extent that the guidelines in 10 CFR 100 could be exceeded. Thus, there are no accidents associated with nearby industrial, transportation, or military facilities, nor associated with on-site stored chemicals that are considered design basis events which require steps to mitigate consequences beyond the design features addressed in the evaluations summarized above, e.g., separation distances.

Section 2.2 References

2.2-201 Seven Gables Airport, AirNav.com http://www.airnav.com/airport/2VG7 accessed October 20, 2007

2.2-202 Federal Aviation Administration, FAA Sectional Aeronautical Charts - Washington North and Washington South, 82nd Edition, Volume 0711, October 2007

2.2-203 U.S. Department of the Navy, Office of the Chief of Naval Operations, Washington, D.C., Letter from S. G. Riley, Captain, to Mr. Marvin Smith, Dominion Resources Services, Inc., Glen Allen, VA., June 8, 2007, 5720, Ser N885F/7U181687

2.2-204 FM Global Guidelines for Evaluating the Effects of Vapor Cloud Explosions Using a TNT Equivalency Method, Data Sheet 7-42

2.2-205 National Oceanic and Atmospheric Administration, Areal Locations of Hazardous Atmospheres (ALOHA®), Version 5.4.1, February 2007

2.2-206 NUREG-1805, Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program

2.2-207 NFPA, “Guide for Aircraft Accident/Incident Response Assessment,” B6.3 and B6.4, 2004 Edition.

2.2-208 United States Coast Guard, Commandant Instruction 16465.12C, 1999. “Chemical Hazards Response Information System, Hazard Chemical Data Manual.”

2.2-209 NALCO Company, Material Safety Data Sheets, H-130 issued December 30, 2005, 3D TRASAR® 3DT177 - issued February 14, 2007, and 3D TRASAR® 3DT104 - issued February 15, 2007.

2.2-210 Perry, R. H., D. W. Green. (1977) Perry's Chemical Engineer's Handbook (7th Edition) (Table 2-5). McGraw-Hill.

2.2-211 Mallinckrodt Baker, Inc., Material Safety Data Sheets, Sodium Bromide - effective date October 19, 2005, Sodium Bisulfate - effective date March 16, 2006, Trisodium Phosphate -effective date November 10, 2005, Sodium Sulfite - effective date June 16, 2005, Disodium Phosphate - effective date May 9, 2005, Sand - effective date August 2, 2006, and Sodium Carbonate - effective date August 17, 2006.

2.2-212 National Institute for Occupational Safety and Health (NIOSH), Center for Disease Control and Prevention (CDC), November 2007.

2.2-213 Federal Emergency Management Agency, U.S. Department of Transportation and U.S. EPA, Handbook of Chemical Hazard Analysis Procedures, Section 11.3, Bulk Transportation of Hazardous Materials by Highway, 1989.

2.2-214 NUREG/CR-6624, Recommendations for Revision of Regulatory Guide 1.78, U.S. Nuclear Regulatory Commission, November 1999.

2.2-215 Virginia Department of Motor Vehicles, 2006 Virginia Traffic Crash Facts.

NAPS COL 2.0-5-A

Table 2.2-201 Airports Within 15 Miles of the Unit 3 Site Since the SSAR

				Number of Flight Operations			Longest Runway		
Airport	**Type**	**Distance**	**Sector**	**Commercial**	**Total**[(a)]	**kd^2** [(b)]	**Orientation**	**Length**	**Comments**
Seven Gables	Private	7.6 miles	NNW	None	Few	28,880	NNW-SSE	1500 ft	Privately owned and operated. Turf runway. No facilities. 1 single-engine plane, 1 helicopter, 1 ultralight based there.

Source: Reference 2.2-201

a. Year 2007

b. RG 1.206: d < 10 miles, k = 500; d >10 miles, k = 1000; where d is the distance in miles from the site, and k is a constant.

NAPS ESP COL 2.2-2

Table 2.2-202 North Anna Unit 3 Onsite Chemical Storage Locations and Quantities

Chemical/Material (Formula/Trade/State)	Location	No. × Quantity (Tank or Tote)
Sodium Hydroxide, NaOH 25% Solution	Water Treatment Building (Inside)	1 × 180 gallon (681 liters) Tote
Alum, 48% Solution (Floculant)	Water Treatment Building (Inside)	1 × 300 gallon (1136 liters) Tote
Sodium Hypochlorite 12% Solution	Hybrid Cooling Tower (Adjacent)	1 × 15,870 gallon (60 m^3) Tank Usable Volume
	Station Water Intake (Unit 3 intake bay)	1 × 2113 gallon (8 m^3) Usable Volume
	Adjacent to Unit 3 Sewage Treatment Plant	2 × 330 gallon (1249 liters) Tote
	Plant Service Water Pump House (Inside)	1 × 1057 gallon (4 m^3) Tank Usable Volume
Nalco 3D TRASAR® 3DT177 (Scale/corrosion Inhibitor) (or equivalent)	Hybrid Cooling Tower (Adjacent)	1 × 1056 gallon (4 m^3) Tank or multiple Totes Usable Volume
	Plant Service Water Pump House (Inside)	1 × 300 gallon (1136 liters) Tote
	Water Treatment Building (Inside)	1 × 55 gallon (208 liters) drum
Nalco 3D TRASAR® 3DT104 (Dispersant) (or equivalent)	Hybrid Cooling Tower (Adjacent)	1 × 5812 gallon (22 m^3) Tank Usable Volume
	Plant Service Water Pump House (Inside)	1 × 400 gallon (1514 liters) Tote
Sodium Bromide (44.7% Solution)	Hybrid Cooling Tower (Adjacent)	1 × 2378 gallon (9 m^3) Tank Usable Volume
	Plant Service Water Pump House (Inside)	1 × 300 gallon (1136 liters) Tote
Nalco H-130, Non-Oxidizing Biocide (or equivalent)	Hybrid Cooling Tower (Adjacent)	3 × 400 gallon (1514 liters) Tote
	Plant Service Water Pump House (Inside)	1 × 300 gallon (1136 liters) Tote
Hydrogen Peroxide 35% Solution	Water Treatment Building (Inside)	1 × 300 gallon (1136 liters) Tote

NAPS ESP COL 2.2-2

Table 2.2-202 North Anna Unit 3 Onsite Chemical Storage Locations and Quantities

Chemical/Material (Formula/Trade/State)	Location	No. × Quantity (Tank or Tote)
Sodium Bicarbonate 12% solution (Prepared from dry chemical powder)	Water Treatment Building (Inside)	1 × 200 gallon (757 liters) Mixing Tank
Sodium Bisulfate 10% solution (Prepared from dry chemical powder)	Plant Service Water Pump House (Inside)	1 × 1056 gallon (4 m^3) Tank Usable Volume
Carbon Dioxide	CO_2 Storage Area- Outside the Turbine Building (West side)	1 × 800 gallon (3028 liters) Tank (Cryogenic Storage Tank)
Hydrogen	Hydrogen Storage Area- Outside the Turbine Building (West side)	1 × 18,000 gallon (68 m^3) Tank (Cryogenic Storage Tank)
Nitrogen	Nitrogen Storage Area- Outside the Reactor Building (West side)	1 × 25,000 gallon (95 m^3) Tank (Cryogenic Storage Tank)
Trisodium Phosphate (0.72% Solution)	Aux. Boiler Building	1 × 555 gallon (2.1 m^3) Tank
Sodium Sulfite (2.2% Solution)	Aux. Boiler Building	1 × 555 gallon (2.1 m^3) Tank
Disodium Phosphate (0.18% Solution)	Aux. Boiler Building	1 × 555 gallon (2.1 m^3) Tank
Oxygen, Liquid	Hydrogen Storage Area - Outside the Turbine Building (West side)	1 × 9000 gallon (34 m^3) Tank (Cryogenic Storage Tank)
Diesel Fuel	North East of Service Building Operation Support Center	2 × 215,400 gallon (815 m^3) Tank
	Ancillary Diesel Building	2 × 15,000 gallon (56 m^3) Storage Tank 2 × 400 gallon (1.5m^3) Day Tank
Sulfuric Acid	NA	Not required based on historic Lake Anna alkalinity

NAPS ESP COL 2.2-2

Table 2.2-202 North Anna Unit 3 Onsite Chemical Storage Locations and Quantities

Chemical/Material (Formula/Trade/State)	Location	No. × Quantity (Tank or Tote)
Urea (Dry Power aqua solution 40% $(NH_2)_2CO$)	Outside the Diesel Generator Building	2 × 12,800 gallon (48 m^3) Tank

NAPS ESP COL 2.2-2

Table 2.2-203 North Anna Unit 3 On-Site Chemicals, Disposition

Chemical/ Chemical Product*	Toxicity Limit (IDLH)	Flammable/ Explosive?	Vapor Pressure	Disposition
Unit 3 Chemicals				
Sodium Hydroxide, NaOH, 25% Solution	None established	No/No	Not required	No further analysis required.
Alum, 48% Solution	None established	No/No	Not required	No further analysis required.
Sodium Hypochlorite, 12% solution	10 ppm for Chlorine	No/No	Not required	Toxicity analysis in Section 6.4. No other analysis required.
Nalco 3D TRASAR® 3DT177 (Scale/corrosion Inhibitor)	1000 mg/m^3 as phosphoric acid	No/No	23.8 mm Hg @25°C	Toxicity analysis in Section 6.4. No other analysis required
Nalco 3D TRASAR® 3DT104 (Dispersant)	None established	No/No	Not required	No further analysis required
Sodium Bromide, 44.7% Solution	None established	No/No	Not required	No further analysis required
Nalco H-130, Non-Oxidizing Biocide	3,300 ppm as ethanol	Yes (3.3–19%) /Yes	30 mm Hg @25°C	Toxicity analysis in Section 6.4. ALOHA and explosion analyses safe separation distances are provided in Table 2.2-204.
Hydrogen Peroxide, 35% Solution	75 ppm	No/No	5 mm Hg @86°F	Toxicity analysis in Section 6.4. No other analysis required.

NAPS ESP COL 2.2-2

Table 2.2-203 North Anna Unit 3 On-Site Chemicals, Disposition

Chemical/ Chemical Product*	Toxicity Limit (IDLH)	Flammable/ Explosive?	Vapor Pressure	Disposition
Unit 3 Chemicals (continued)				
Sodium Bicarbonate, 12% Solution (prepared from dry chemical powder)	None established	No/No	Not required	No further analysis required.
Sodium Bisulfate, 10% Solution (prepared from dry chemical powder)	None established	No/No	Not required	No further analysis required.
Carbon Dioxide (Cryogenic Storage Tank)	40,000 ppm	No/No	907.299 psi @75°F	Toxicity (asphyxiation) analysis in Section 6.4, no other analysis required.
Hydrogen, Gas	None established; Asphyxiant	Yes (4–75%)/ Yes	29.030 @–418°F	Toxicity (asphyxiation) analysis in Section 6.4. ALOHA and explosion analyses safe separation distances are provided in Table 2.2-204.
Nitrogen, Gas	None established; Asphyxiant	No/No	65.820 @–294°F	Toxicity (asphyxiation) analysis in Section 6.4. No other analysis required.
Trisodium Phosphate, 0.72% Solution	None established	No/No	Not required	No further analysis required.
Sodium Sulfite, 2.2% Solution	None established	No/No	17.535 mm Hg @93.6°F	No further analysis required.

NAPS ESP COL 2.2-2

Table 2.2-203 North Anna Unit 3 On-Site Chemicals, Disposition

Chemical/ Chemical Product*	Toxicity Limit (IDLH)	Flammable/ Explosive?	Vapor Pressure	Disposition
Unit 3 Chemicals (continued)				
Disodium Phosphate, 0.18% Solution	None established	No/No	Not required	No further analysis required.
Oxygen, Gas	None established; asphyxiant	No/No	36.260 psi @−280°F	Toxicity (asphyxiation) analysis in Section 6.4. No other analysis required.
Diesel Fuel (Unit 3)	None established	Yes (1.3–6.0%)/ No	0.100 psi @100°F	No further analysis is required[1,2].
Sulfuric Acid	NA	NA	Not required	Not required based on historic Lake Anna alkalinity
Urea, $(NH_2)_2CO$ 40% Solution (prepared from dry powder)	None established	No/No	Not required	No further analysis required.

* Properties confirmed by Material Safety Data Sheets (References 2.2-208, 2.2-209, 2.2-210, 2.2-211, and 2.2-212).

1. Chemicals with vapor pressures less than 10 torr (0.193 psi) were not considered significant hazards since at these vapor pressures the chemicals are not very volatile. Under normal conditions, these chemicals do not enter the atmosphere fast enough to reach concentrations hazardous to people and, therefore, are not considered to be an air dispersion hazard. (Reference 2.2-205)
2. A fluid with an extremely low vapor pressure will not explode per NFPA 422 (Reference 2.2-207) which states that the vapor space in tanks storing low vapor pressure liquids is normally too lean to burn. The vapor pressure of diesel fuel is low enough such that the vapor concentration above the liquid (0.36%) is significantly lower than the LFL (1.3%). As a result the air-gas mixture is expected to be too lean to ignite and/or explode.

NAPS ESP COL 2.2-2

Table 2.2-204 Design Basis Events, Explosions, Flammable Vapor Clouds (Delayed Ignition) and Vapor Cloud Explosions

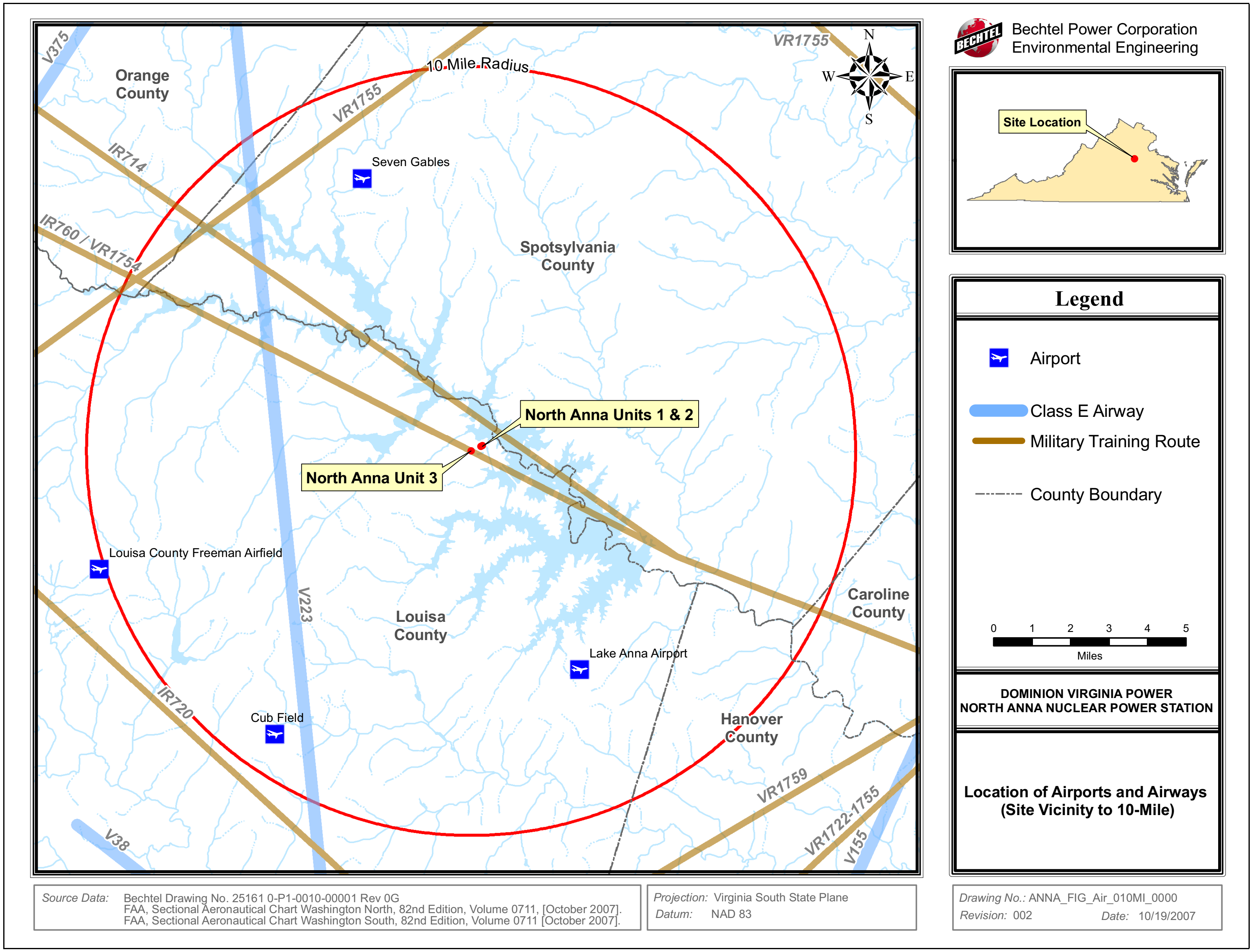
Chemical Evaluated	Quantity (gallons)	Distance to Nearest Safety Related Structure for Unit 3 (ft)	Distance for Explosion to have less than 1 psi of Peak Incident Pressure (ft)[(a)]	Distance to Lower Flammability Limit (ft)[(b)]	Safe Distance for Vapor Cloud Explosions (ft)[(c)]
Nalco H-130©	400 (1514 liters)	1,402 (427 m)	86 (26 m)	<33 (<10 m)	72 (22 m)
Hydrogen	18,000 ($68\ m^3$)	752 (229 m)	273 (83 m)	222 (68 m)	258 (79 m)

(a) The minimum separation distance required for an in-vessel confined explosion to have less than 1 psi peak incident pressure.

(b) The distance from the spill site where the vapor cloud can exist and still be between the upper and lower flammability limit, presenting the possibility of ignition.

(c) The distance from the spill site to the location where the pressure wave from the detonation of the traveled vapor cloud is at 1 psi overpressure.

Figure 2.2-201 Civilian and Military Airway Routes in NAPS Vicinity



2.3 Meteorology

2.3.1 Regional Climatology

NAPS COL 2.0-7-A

The information needed to address the DCD COL Item 2.0-7-A is included in SSAR Section 2.3.1, which is incorporated by reference with the following supplement.

2.3.1.2 General Climate

This SSAR section is supplemented by inserting, as the third paragraph, the following information about temperature extremes.

NAPS COL 2.0-7-A

Using the International Station Meteorological Climate Summary for Richmond (Reference 2.3-207), dry-bulb temperatures ranging from -31.6ºC (-25ºF) to 38.3ºC (101ºF), were plotted in 1.1ºC (2ºF) intervals with their maximum observed coincident wet-bulb temperatures to obtain a corresponding curve. Extrapolating the curve to 42.8ºC (109ºF), which is the 100-year return value for maximum dry-bulb temperature, the 100-year return value for coincident wet-bulb temperature was determined to be 24.4ºC (76ºF). That is, 24.4ºC (76ºF) is the coincident wet-bulb temperature corresponding to the 100-year return period value for maximum dry-bulb temperature.

2.3.1.3.1 Extreme Winds

This SSAR section is supplemented with information to address wind speeds used for part of the Unit 3 design as follows.

NAPS COL 2.0-7-A

Nonsafety-related structures, not included as part of the certified design, are designed in accordance with Part I of the Virginia Uniform Statewide Building Code (Reference 2.3-204), which incorporates by reference the International Building Code (IBC) (Reference 2.3-205). The applicable edition of the IBC invokes Section 6 of American Society of Civil Engineers (ASCE) Standard No. 7 (Reference 2.3-206). ASCE 7, Section 6.5.4, Figure 6.1, defines the basic wind speed for such structures. Unit 3 is not in a Special Wind Region.

The basic wind speed for Unit 3 nonsafety-related structures, not included in the certified design, is 40 m/s (90 mph). This design value is defined in Reference 2.3-206 as a 3-second gust at 10 m (33 ft) above the ground that has a 2 percent annual probability of being exceeded (i.e., the 50-year mean recurrence interval).

2.3.1.3.4 Precipitation Extremes

The last paragraph in this SSAR section is supplemented as follows with information to address ice and winter precipitation for Unit 3 safety-related structures.

NAPS COL 2.0-7-A

As Section 2.4.7.6 indicates, the design features that demonstrate acceptable roof structure performance are described in DCD Appendix 3G, e.g., for the reactor building, see DCD Section 3G.1.5.

2.3.2 Local Meteorology

NAPS COL 2.0-8-A

The information needed to address the DCD COL Item 2.0-8-A is included in SSAR Section 2.3.2, which is incorporated by reference with the following supplements.

2.3.2.3 Potential Influence of the Plant and the Facilities on Local Meteorology

The fourth paragraph of this SSAR section is revised as follows with information to address the impacts of cooling tower operations.

NAPS COL 2.0-8-A

The convective and conductive heat losses to the atmosphere resulting from the operation of the Unit 3 closed cycle, hybrid and dry cooling tower system dissipate rapidly through continuous mixing with the surrounding moving air mass. Therefore, any increase in overall ambient temperature is very localized to the NAPS site and does not affect the ambient atmospheric and ground temperature beyond the NAPS site.

The sixth paragraph of this SSAR section is revised to address the engineering performed to consider potential impacts of Unit 3 cooling tower operations as follows.

NAPS ESP COL 2.3-1

The impact on the design and operation of Unit 3 from any cooling-tower-induced increase in the local ambient air temperature, or moisture and salt content, has been considered in the location and separation of wet cooling towers relative to electrical transmission lines and electrical equipment, including transformers and switchyard. Also, the separation of the wet and dry towers from Unit 3 buildings considered potential effects on air ambient conditions at HVAC air intakes, including consideration of prevailing winds. The site layout shown in

Figure 2.1-201 ensures minimal impacts on Unit 3 operation from local increases in ambient air temperature, moisture content, and moisture and salt deposition resulting from the operation of the Unit 3 cooling towers, including wet cooling tower drift and plume condensation.

2.3.2.3.1 Salt Deposition and Moisture

The potential impacts on Unit 3 plant design and operation due to salt deposition, fogging, and icing from the CIRC hybrid cooling tower and from the Plant Service Water System (PSWS) cooling tower were assessed using the Seasonal/Annual Cooling Tower Impact (SACTI) computer code (Reference 2.3-202). See Section 10.4.5.8 for further description of the hybrid cooling tower design and see Section 9.2.1.2 for the service water cooling tower design.

a. **Salt Deposition**

The service water cooling tower produces higher salt deposition rates than the CIRC hybrid cooling tower even though the CIRC hybrid cooling tower is modeled with a higher drift rate of 0.001 percent. Therefore, only the limiting SACTI analysis for the effects of salt deposition from the service water cooling tower on the Unit 3 electrical transformers is discussed below. The following assumptions were made in the SACTI model for the service water cooling tower:

- Drift loss is 0.0005 percent.
- Total dissolved solids concentration of the cooling water is 9.0×10^{-4} g salt/cm^3.
- Salt density is 2.17 g/cm^3.

Salt deposition from evaporative cooling towers has the potential to build up on bushings of electrical equipment such as Unit 3 transformers, switchyard equipment, and transmission lines (see Figure 8.2-202). A highest deposition rate of 0.0216 mg/cm^2-month is predicted to occur near the Unit 3 transformers during the summer season. The transmission lines and switchyard have lower predicted maximum deposition rates than the transformers. Several months of buildup at this rate would be needed before such deposits would accumulate to 0.08 mg/cm^2, which is the upper end of the “Light Contamination Level” range defined by the applicable IEEE standard (Reference 2.3-203). However, due to the service water cooling tower location with respect to prevailing wind directions, and natural wash off from local precipitation, total deposits are not expected to reach a level requiring attention.

Therefore, cooling tower plume generated salt deposits are not expected to adversely affect any electrical equipment at the North Anna Site.

b. **Moisture**

Added humidity and potential moisture impacts due to CIRC hybrid cooling tower and service water cooling tower operation are predicted by the hours of fogging and icing produced by each tower as determined in the SACTI analysis. The following assumptions were used in the analysis:

- Plume abatement is not accounted for in the SACTI model.
- Total airflow for wet and dry sections of the CIRC hybrid cooling tower is considered.
- The CIRC hybrid cooling tower is modeled as one cell with a combined flow rate of all fans.

A maximum of 9.5 hours of fogging per year at any location due to cooling tower operation is predicted for both the CIRC hybrid cooling tower and service water cooling tower. Because the HVAC intakes, onsite transmission lines, switchyard equipment, and transformers are designed for outdoor operations, which include environmental conditions such as rain, fog and snow, added fog and moisture from cooling tower plumes are not expected to have an adverse affect on these plant features. Both cooling towers incorporate plume-limiting technology; therefore, the predicted annual hours of fogging due to cooling tower operation are conservative. Additionally, the SACTI analysis predicts no icing will occur.

2.3.2.3.2 **Ambient Air Temperature Increases**

In addition to the CIRC hybrid cooling tower and service water cooling tower, the CIRC dry cooling tower was considered when evaluating the potential for local ambient air temperature increases. The evaluation was based on the following assumptions:

- CIRC hybrid cooling tower height is 55 m (180 ft).
- CIRC dry cooling tower height is 19.8 m (65 ft).
- Service water cooling tower height is 18.5 m (61 ft).
- The highest control room HVAC air intakes height is approximately 8 m (26.2 ft).
- Exhaust plume temperatures of the CIRC hybrid and dry cooling towers are no greater than the maximum inlet water temperature of 51.6°C (125°F).

- Exhaust plume temperature of the service water cooling tower is no greater than the maximum inlet water temperature of 39°C (103°F).

The Unit 3 site characteristic 0 percent exceedance value for ambient design temperature is 40.5°C (104.9°F) dry bulb. As shown in DCD Table 3.2-1, the control building HVAC system is classified as Safety Class 3 and is the only HVAC system with safety class components, other than isolation equipment. Operation of the control building HVAC system maintains the control room habitability area (CRHA) within the temperature and relative humidity ranges in DCD Table 9.4-1, which shows the limiting outside air design condition temperature for the control room HVAC intakes is 47.2°C (117°F) dry bulb.

A cooling tower plume would need to raise the local ambient temperature associated with the surrounding air mass at the control room HVAC intakes by more than 6.7°C (12.1°F) to exceed the design value. However, cooling tower plume temperatures are higher than the local ambient air temperatures, so buoyancy causes the thermal plume to rise under low wind conditions; whereas, high wind conditions that could direct a plume towards the intakes, would result in rapid air dispersion and mixing that cools the plume. Because the Unit 3 control room HVAC intakes are at a lower elevation than the exhaust plenums of the CIRC hybrid and dry cooling towers, and because the control room HVAC intakes are located approximately 500 m (1640 ft) from the CIRC towers, the thermal plumes from the towers are not expected to raise the local ambient air temperatures at intakes for the control room HVAC systems above the design value. The maximum inlet water temperature of 39°C (103°F) for the service water cooling tower is lower than the limiting outside air design condition temperature of 47.2°C (117°F) for the control room HVAC systems. Therefore, exhaust from the service water cooling tower will not adversely affect the control room HVAC systems due to increases in surrounding ambient air temperature.

Similarly, the exhausts from the cooling towers are not expected to affect local ambient air temperatures near Unit 3 electrical equipment, including the transformers and switchyard equipment, which are at lower elevations than the Unit 3 main control room HVAC intakes. As with the HVAC intakes, high wind conditions that could direct a plume towards the outdoor electrical equipment would result in rapid air dispersion and mixing that cools the plume. Therefore, exhausts from the cooling towers will not adversely affect such Unit 3 electrical equipment due to increases in surrounding ambient air temperature.

2.3.3 Onsite Meteorological Measurements Program

NAPS COL 2.0-9-A

The information needed to address the DCD COL Item 2.0-9-A is included in SSAR Section 2.3.3, which is incorporated by reference with the following supplement.

2.3.3.1.2 Location, Elevation, and Exposure of Instruments

The second paragraph of this SSAR section is supplemented as follows with information to address the acceptability of distances from Unit 3 to the wind measurement towers.

NAPS COL 2.0-9-A

The highest building at the Unit 3 site is the Turbine Building at 57.9 m (190 ft) above design plant grade level of 88.4 m (290 ft). The primary meteorological measurements tower is located about 733.4 m (2406 ft) east of the plant facility boundary. Since the primary tower is located more than 10 building heights away from the tallest building at the Unit 3 site, the Unit 3 turbine building does not influence the meteorological measurements. The backup meteorological tower is located about 744 m (2440 ft) away from the highest building. Therefore, the turbine building also does not influence the meteorological measurements taken at the backup meteorological measurements tower.

NAPS COL 2.0-10-A

2.3.4 Short-Term (Accident) Diffusion Estimates

The information needed to address the DCD COL Item 2.0-10-A is included in SSAR Section 2.3.4, which is incorporated by reference with the following supplements.

2.3.4.1 Basis

The eighth paragraph of this SSAR section is supplemented as follows with information to address the wake influence zone of tall buildings at the Unit 3 site.

NAPS COL 2.0-10-A

As described in SSAR Section 2.1, the EAB is the perimeter of a 5000-foot-radius circle from the center of the containment of the third of the four originally proposed units. The highest building at the Unit 3 site is the Turbine Building which is 57.9 m (190 ft) above design plant grade level. Therefore, the closest point on the EAB is more than 10 building heights away from the Unit 3 power block buildings which could have postulated fission product releases. As a result, the entire EAB is located beyond the wake influence zone that can be induced by tall buildings, e.g., the Unit 3 Turbine Building or Reactor Building.

NAPS ESP COL 2.3-2

2.3.4.3 Atmospheric Dispersion Factors for On-Site Doses

Onsite χ/Q values for use in evaluating potential doses from Unit 3 postulated release locations (sources) to on-site receptor locations are based on the Unit 3 plant layout shown in DCD Figure 2A-1. The meteorological data used in evaluating on-site doses is the same data used for the accident condition dose calculations in SSAR Section 2.3.4. The χ/Q values for the control room and technical support center were calculated using the ARCON96 computer code in accordance with guidance as documented in RG 1.194. The source and receptor combinations are shown in Table 2.3-201 through Table 2.3-207. DCD Figure 2A-1 shows the locations of postulated accidental releases from Unit 3 and the Unit 3 receptor locations.

NAPS COL 2.0-11-A

2.3.5 Long-Term (Routine) Diffusion Estimates

The information needed to address DCD COL Item 2.0-11-A is included in SSAR Section 2.3.5, which is incorporated by reference with the following supplements and variances.

2.3.5.1 Basis

The third through sixth paragraphs of this SSAR section are supplemented as follows with information to address the receptors near the Unit 3 site.

NAPS ESP COL 2.3-3

The following input data and assumptions were used in the XOQDOQ modeling:

- Meteorological Data: Three-year combined (1996–1998) onsite joint frequency distribution of wind speed, wind direction, and atmospheric stability.
- Type of Release: Ground level.
- Wind Sensor Height: 10 m (33 ft).
- Vertical Temperature Difference: 10 m (33 ft) – 48.4 m (158.9 ft).
- Number of Wind Speed Categories: 7.

- Release Height: 10 m (33 ft) (default height).
- Reactor Building Height: 49 m (161 ft).
- Minimum Reactor Building Cross-Sectional Area: 2400 m^2 (25,800 ft^2).
- Distances from the release point to the nearest residence, nearest site boundary, milk cow, vegetable garden, milk goat, meat animal: See Table 2.3-15R.

For the dispersion analysis, the ESBWR Reactor Building is used to determine the minimum building cross-sectional area for evaluating building downwash effects. The height of this building is approximately 49 m (161 ft) including parapets. Based on this height and a nominal width of 49 m (161 ft) on the rectangular face of the building, a minimum building cross-sectional area of 2400 m^2 (25,800 ft^2) was used to determine χ/Q and D/Q estimates. The perpendicular face of the building is narrower at the top, but the total area, including stairwells and the elevator shaft, is greater than 2400 m^2 (25,800 ft^2) in that perpendicular direction. For the NAPS site, the χ/Q and D/Q values were found to depend on building height but not cross-sectional area.

The annual Radiological Environmental Monitoring Program (Reference 2.3-201) was reviewed to determine if the distances of any of the nearest receptors modeled for the SSAR have changed. The results are documented in Table 2.3-15R based on a subsequent review and plotting of receptor locations using Geographic Information System (GIS) technology. This process provided improved distance accuracy for these receptors. The results show the closest receptor to be a residence in the NW direction at a distance of 1.36 km (4453 ft). The evaluation assumed conservatively, that each receptor (meat animal, vegetable garden, residence) is at the location of the closest receptor and that the closest receptor is the residence in the NW direction at the previously determined distance of 1.20 km (3930 ft). Therefore, for the purposes of the atmospheric dispersion analysis and the subsequent dose evaluations, one of each type of receptor was assumed to be at 1.20 km (3930 ft) in each compass direction. The maximum annual average χ/Q value calculated for the nearest residence, vegetable garden, and meat animal, all assumed at 1.20 km (3930 ft), is 4.20 E-6 sec/m^3 in the ESE direction. The maximum D/Q for these receptors is 1.10E-8 m^{-2} in the NNE direction. In the evaluation performed for this FSAR, the shortest distance from any point on the plant facility boundary to the site boundary

(EAB) was found to be 1.6 km (1.0 mile) in the direction where the maximum χ/Q is calculated. However, for conservatism, the greater χ/Q from SSAR Section 2.3.5, which is based on a distance of 1.42 km (0.88 miles), is retained for use in this section. The maximum annual χ/Q (no decay, undepleted) at the EAB is 3.70×10^{-6} sec/m^3; at a distance of 1.42 km (0.88 mile) to the ESE of the plant facility boundary (Figure 2.0-205).

NAPS ESP VAR 2.0-1a to 2.0-1l

The results are summarized in Table 2.3-16R and Table 2.3-17R. These tables present the maximum calculated χ/Qs and D/Qs at receptors and at various distances from the site.

Add the following at the end of this SSAR section to address annual average χ/Q and D/Q estimates.

NAPS COL 2.0-11-A

Long-term (annual average) χ/Q and D/Q estimates generated by the XOQDOQ model are also presented for each directional sector at twenty-two specific distances, as well as for ten distance segments. Table 2.3-208 presents the no decay and undepleted χ/Q estimates at various downwind distances between 0.4 km (0.25 mi) and 80.5 km (50 mi). Table 2.3-209 presents the no decay and undepleted χ/Q estimates for various distance segments out to 80.5 km (50 mi).

Table 2.3-210 presents the 2.26 day decay (for short-lived noble gases) and undepleted χ/Q estimates at the same downwind distances. Table 2.3-211 presents the 2.26 day decay and undepleted χ/Q estimates for the same distance segments.

Table 2.3-212 presents the 8 day decay (for all iodines released to the atmosphere) and depleted χ/Q estimates at the same downwind distances. Table 2.3-213 presents the 8 day decay and depleted χ/Q estimates for the same distance segments.

Table 2.3-214 presents the D/Q estimates for the same downwind distances. Table 2.3-215 presents the D/Q estimates for the same distance segments.

Section 2.3 References

2.3-201 Dominion North Anna Power Station 2006 Annual Radiological Environmental Operating Report, prepared by Dominion North Anna Power Station, January 2006-December 2006.

2.3-202 SACTI User’s Manual: Cooling-Tower-Plume Prediction Code, EPRI CS-3403-CCM, April 1984.

2.3-203 Institute of Electrical and Electronics Engineers, Std C57.19.100, “IEEE Guide for Application of Power Apparatus Bushings.”

2.3-204 Virginia Uniform Statewide Building Code, Part I (Virginia Construction Code), Virginia Board of Housing and Community Development.

2.3-205 International Building Code, International Code Council, Inc.

2.3-206 Minimum Design Loads for Buildings and Other Structures, American Society of Civil Engineers Standard No. 7 (ASCE 7).

2.3-207 International Station Meteorological Climate Summary, Fleet Numerical Meteorology and Oceanography Detachment, National Climatic Data Center, and USAFETAC OL-A, Version 4.0, September 1996.

NAPS ESP COL 2.3-3

Table 2.3-15R Source to Receptor Distances

Type[3]	Direction from Unit 3	Distance from Plant Facility Boundary (ft)[1]	Distance from Plant Facility Boundary (miles/km)[1]
Vegetation			
Veg	S	5546	1.05/1.69
Veg	SSW	No Receptor	
Veg	SW	17268	3.27/5.26
Veg	WSW	11021	2.09/3.36
Veg	W	No Receptor	
Veg	WNW	7895	1.50/2.41
Veg	NW	No Receptor	
Veg	NNW	4765	0.90/1.45
Veg	N	5891	1.12/1.80
Veg	NNE	17164	3.25/5.23
Veg	NE	5284	1.00/1.61
Veg	ENE	13230	2.51/4.03
Veg	E	9281	1.76/2.83
Veg	ESE	No Receptor	
Veg	SE	4663	0.88/1.42
Veg	SSE	4669	0.88/1.42
Meat Animal			
Meat	S	13483	2.55/4.11
Meat	SSW	7877	1.49/2.40
Meat	SW	No Receptor	
Meat	WSW	5769	1.09/1.76
Meat	W	No Receptor	
Meat	WNW	18697	3.54/5.70
Meat	NW	No Receptor	
Meat	NNW	No Receptor	

NAPS ESP COL 2.3-3

Table 2.3-15R Source to Receptor Distances

Type[3]	Direction from Unit 3	Distance from Plant Facility Boundary (ft)[1]	Distance from Plant Facility Boundary (miles/km)[1]
Meat Animal (continued)			
Meat	N	No Receptor	
Meat	NNE	8573	1.62/2.61
Meat	NE	8357	1.58/2.55
Meat	ENE	13738	2.60/4.19
Meat	E	19588	3.71/5.97
Meat	ESE	No Receptor	
Meat	SE	8023	1.52/2.45
Meat	SSE	14210	2.69/4.33
Resident			
Res	S	4718	0.89/1.44
Res	SSW	5853	1.11/1.78
Res	SW	6513	1.23/1.99
Res	WSW	No Receptor	
Res	W	No Receptor	
Res	WNW	5802	1.10/1.77
Res	NW	3930	0.74/1.20[2]
Res	NNW	4565	0.86/1.39
Res	N	4949	0.94/1.51
Res	NNE	8194	1.55/2.50
Res	NE	4926	0.93/1.50
Res	ENE	12348	2.34/3.76
Res	E	7981	1.51/2.43
Res	ESE	No Receptor	
Res	SE	4832	0.92/1.47
Res	SSE	No Receptor	

NAPS ESP COL 2.3-3

Table 2.3-15R Source to Receptor Distances

Type[3]	Direction from Unit 3	Distance from Plant Facility Boundary (ft)[1]	Distance from Plant Facility Boundary (miles/km)[1]
Site Boundary (Exclusion Area Boundary)			
EAB	S	3719	0.70/1.13
EAB	SSW	3238	0.61/0.99
EAB	SW	2877	0.54/0.88
EAB	WSW	2891	0.55/0.88
EAB	W	2914	0.55/0.89
EAB	WNW	3393	0.64/1.03
EAB	NW	3919	0.74/1.19
EAB	NNW	4417	0.84/1.35
EAB	N	4847	0.92/1.48
EAB	NNE	5110	0.97/1.56
EAB	NE	4858	0.92/1.48
EAB	ENE	4967	0.94/1.51
EAB	E	5604	1.06/1.71
EAB	ESE	5304	1.00/1.62
EAB	SE	4603	0.87/1.40
EAB	SSE	4180	0.79/1.27

Notes:
1. Distances are from the plant facility boundary. See Figure 2.0-205.
2. Actual distance is 1.36 km (4453 ft).
3. No milk cows or goats within a 5-mile radius of NAPS.

NAPS ESP COL 2.3-3
NAPS ESP VAR 2.0-1a to 2.0-1l

Table 2.3-16R XOQDOQ Predicted Maximum χ/Q and D/Q Values at Specific Points of Interest

Type of Location	Direction from Site	Distance (miles)	χ/Q (No Decay, Undepleted)	χ/Q (2.26 Day Decay, Undepleted)	χ/Q (8 Day Decay, Depleted)	D/Q
Residence	ESE	0.74	4.20E-06	4.10E-06	3.70E-06	1.1E-08[b]
EAB[c]	ESE	0.88	3.7E-06	3.7E-06	3.3E-06	1.2E-08[a]
Meat Animal	ESE	0.74	4.20E-06	4.10E-06	3.70E-06	1.1E-08[b]
Veg. Garden	ESE	0.74	4.20E-06	4.10E-06	3.70E-06	1.1E-08[b]

Notes:

χ/Q – sec/m^3

D/Q – 1/m^2

a: direction South and distance of 0.62 mi for maximum D/Q for EAB

b: direction North-Northeast for maximum D/Q for residence, meat animal, and vegetable garden

c: from SSAR Table 2.3-16

NAPS ESP COL 2.3-3

Table 2.3-17R XOQDOQ Predicted Maximum Annual Averages (Ground-Level Release)

No Decay Undepleted

	Distance In Miles from Site										
ESE	0.25	0.5	0.75	1	1.5	2	2.5	3	3.5	4	4.5
X/Q (s/m^3)	2.566E-05	7.927E-06	4.114E-06	2.670E-06	1.524E-06	1.038E-06	7.709E-07	6.052E-07	4.936E-07	4.140E-07	3.546E-07
	Distance In Miles from Site										
ESE	5	7.5	10	15	20	25	30	35	40	45	50
X/Q (s/m^3)	3.089E-07	1.823E-07	1.258E-07	7.493E-08	5.206E-08	3.932E-08	3.130E-08	2.583E-08	2.188E-08	1.891E-08	1.660E-08
	Segment Boundaries In Miles from Site										
ESE	0.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50	
X/Q (s/m^3)	4.319E-06	1.563E-06	7.757E-07	4.952E-07	3.553E-07	1.853E-07	7.606E-08	3.951E-08	2.588E-08	1.893E-08	

2.26 Day Decay Undepleted

	Distance In Miles from Site										
ESE	0.25	0.5	0.75	1	1.5	2	2.5	3	3.5	4	4.5
X/Q (s/m^3)	2.562E-05	7.901E-06	4.094E-06	2.653E-06	1.509E-06	1.024E-06	7.584E-07	5.935E-07	4.825E-07	4.033E-07	3.443E-07
	Distance In Miles from Site										
ESE	5	7.5	10	15	20	25	30	35	40	45	50
X/Q (s/m^3)	2.989E-07	1.735E-07	1.178E-07	6.789E-08	4.566E-08	3.339E-08	2.573E-08	2.057E-08	1.688E-08	1.413E-08	1.202E-08

NAPS ESP
COL 2.3-3

Table 2.3-17R XOQDOQ Predicted Maximum Annual Averages (Ground-Level Release)

	Segment Boundaries In Miles from Site									
ESE	0.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
X/Q (s/m^3)	4.300E-06	1.548E-06	7.634E-07	4.840E-07	3.450E-07	1.766E-07	6.909E-08	3.360E-08	2.064E-08	1.416E-08

8 Day Decay Depleted	Distance In Miles from Site										
ESE	0.25	0.5	0.75	1	1.5	2	2.5	3	3.5	4	4.5
X/Q (s/m^3)	2.428E-05	7.232E-06	3.661E-06	2.333E-06	1.291E-06	8.561E-07	6.216E-07	4.781E-07	3.827E-07	3.154E-07	2.659E-07
	Distance In Miles from Site										
ESE	5	7.5	10	15	20	25	30	35	40	45	50
X/Q (s/m^3)	2.281E-07	1.267E-07	8.293E-08	4.530E-08	2.928E-08	2.076E-08	1.560E-08	1.221E-08	9.839E-09	8.111E-09	6.808E-09
	Segment Boundaries In Miles from Site										
ESE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50	
X/Q (s/m^3)	3.864E-06	1.329E-06	6.267E-07	3.843E-07	2.666E-07	1.298E-07	4.654E-08	2.097E-08	1.227E-08	8.140E-09	

Relative Deposition	Distance In Miles from Site										
NNE	0.25	0.5	0.75	1	1.5	2	2.5	3	3.5	4	4.5
D/Q (1/m^2)	6.257E-08	2.116E-08	1.086E-08	6.671E-09	3.326E-09	2.017E-09	1.364E-09	9.882E-10	7.514E-10	5.920E-10	4.793E-10
	Distance In Miles from Site										
NNE	5	7.5	10	15	20	25	30	35	40	45	50
D/Q (1/m^2)	3.964E-10	1.943E-10	1.219E-10	6.161E-11	3.729E-11	2.500E-11	1.792E-11	1.345E-11	1.046E-11	8.355E-12	6.820E-12

NAPS ESP
COL 2.3-3

Table 2.3-17R XOQDOQ Predicted Maximum Annual Averages (Ground-Level Release)

	Segment Boundaries In Miles from Site									
NNE	0.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
D/Q (1/m^2)	1.129E-08	3.487E-09	1.388E-09	7.583E-10	4.820E-10	2.070E-10	6.420E-11	2.544E-11	1.359E-11	8.410E-12

NAPS ESP COL 2.3-2

Table 2.3-201 Unit 3 Reactor Building χ/Q Results (sec/m^3)

Source/Receptor[1]	0–2 hr	2–8 hr	8–24 hr	1–4 d	4–30 d
RB to CBL[2]	1.74E-03	1.17E-03	4.07E-04	3.42E-04	2.79E-04
RB-VS to CBL[2]	9.08E-04	6.36E-04	2.36E-04	1.72E-04	1.41E-04
RB to EN[3]	1.14E-03	8.18E-04	2.85E-04	2.32E-04	2.02E-04
RB to ES[3]	1.14E-03	8.25E-04	3.11E-04	2.44E-04	2.02E-04
RB to N[3]	1.25E-03	8.88E-04	3.41E-04	2.69E-04	2.20E-04
RB-VS to ES[3]	6.68E-04	4.60E-04	1.72E-04	1.22E-04	1.03E-04
RB-VS to N[3]	7.28E-04	5.03E-04	1.87E-04	1.34E-04	1.13E-04
RB to TSCE[4]	2.32E-04	1.79E-04	7.54E-05	5.85E-05	4.57E-05
RB to TCSW[4]	2.63E-04	2.17E-04	9.35E-05	6.71E-05	5.21E-05

Note 1: See DCD Figure 2A-1 for building source and intake locations.
Note 2: These results are for confirmation of the Reactor Building to Control Room Unfiltered Inleakage χ/Q values.
Note 3: These results are for confirmation of the Reactor Building to Control Room Intake χ/Q values.
Note 4: These results are for confirmation of the Reactor Building to Technical Support Center Intake and Inleakage χ/Q values.

NAPS ESP COL 2.3-2

Table 2.3-202 Unit 3 Turbine Building χ/Q Results (sec/m^3)

Source/Receptor	0–2 hr	2–8 hr	8–24 hr	1–4 d	4–30 d
TB to CBL[1]	6.71E-04	3.42E-04	1.53E-04	1.17E-04	9.19E-05
TB-VS to CBL[1]	3.17E-04	2.60E-04	1.03E-04	7.44E-05	5.61E-05
TB-TD to CBL[1]	2.50E-04	2.21E-04	8.85E-05	5.84E-05	4.47E-05
TB to EN[2]	8.17E-04	3.96E-04	1.78E-04	1.50E-04	1.15E-04
TB to ES[2]	5.96E-04	3.19E-04	1.37E-04	1.11E-04	8.43E-05
TB to N[2]	5.50E-04	2.97E-04	1.29E-04	1.02E-04	7.88E-05
TB-TD to EN[2]	2.42E-04	2.08E-04	8.50E-05	5.65E-05	4.55E-05
TB-VS to EN[2]	3.49E-04	2.91E-04	1.22E-04	8.16E-05	6.84E-05
TB-VS to N[2]	2.66E-04	2.19E-04	9.22E-05	6.14E-05	5.01E-05
TB to TSCE[3]	9.02E-04	5.82E-04	1.98E-04	1.84E-04	1.62E-04
TB to TSCW[3]	2.00E-03	1.13E-03	4.45E-04	3.78E-04	3.27E-04
TB-TD to TSCW[3]	1.13E-03	7.96E-04	3.55E-04	2.41E-04	2.17E-04

Note 1: These results are for confirmation of the Turbine Building to Control Room Unfiltered Inleakage χ/Q values.

Note 2: These results are for confirmation of the Turbine Building to Control Room Intake χ/Q values.

Note 3: These results are for confirmation of the Turbine Building to Technical Support Center Intake and Inleakage χ/Q values.

NAPS ESP COL 2.3-2

Table 2.3-203 Unit 3 Reactor Building Roof/PCCS Vent χ/Q Results (sec/m^3)

Source/Receptor	0–2 hr	2–8 hr	8–24 hr	1–4 d	4–30 d
PCCS to CBL[1]	1.58E-03	1.34E-03	5.61E-04	3.96E-04	3.34E-04
PCCS to EN[2]	1.31E-03	9.35E-04	3.66E-04	2.70E-04	2.18E-04
PCCS to ES[2]	1.07E-03	8.29E-04	3.51E-04	2.55E-04	2.08E-04
PCCS to N[2]	1.08E-03	8.53E-04	3.72E-04	2.59E-04	2.17E-04
PCCS to TSCE[3]	3.44E-04	2.80E-04	1.13E-04	8.58E-05	6.63E-05
PCCS to TSCW[3]	4.40E-04	3.64E-04	1.52E-04	1.16E-04	8.78E-05

Note 1: These results are for confirmation of the Passive Containement Cooling System to Control Room Unfiltered Inleakage χ/Q values.
Note 2: These results are for confirmation of the Passive Containement Cooling System to Control Room Intake χ/Q values.
Note 3: These results are for confirmation of the Passive Containement Cooling System to Technical Support Center Intake and Inleakage χ/Q values.

NAPS ESP COL 2.3-2

Table 2.3-204 Unit 3 Fuel Building χ/Q Results (sec/m^3)

Source/Receptor	0–2 hr	2–8 hr	8–24 hr	1–4 d	4–30 d
FB to CBL[1]	2.62E-03	1.97E-03	7.26E-04	6.01E-04	5.20E-04
FB to EN[2]	1.23E-03	9.40E-04	3.49E-04	2.85E-04	2.44E-04
FB to ES[2]	1.71E-03	1.29E-03	4.68E-04	3.73E-04	3.28E-04
FB to N[2]	2.15E-03	1.59E-03	5.90E-04	4.70E-04	4.02E-04

Note 1: These results are for confirmation of the Fuel Building to Control Room Unfiltered Inleakage χ/Q values.
Note 2: These results are for confirmation of the Fuel Building to Control Room Intake χ/Q values.

NAPS ESP COL 2.3-2

Table 2.3-205 Unit 3 Radwaste Building χ/Q Results (sec/m^3)

Source/Receptor[1]	0–2 hr	2–8 hr	8–24 hr	1–4 d	4–30 d
RW-VS to CBL[2]	6.13E-04	4.90E-04	2.19E-04	1.58E-04	1.29E-04
RW to N[3]	4.61E-04	3.74E-04	1.66E-04	1.16E-04	9.85E-05
RW-VS to EN[3]	4.69E-04	3.76E-04	1.61E-04	1.17E-04	9.96E-05
RW-VS to N[3]	4.17E-04	3.29E-04	1.47E-04	1.06E-04	8.60E-05

Note 1: The PCCS vent χ/Q values are assumed to bound the χ/Q values for any relaease from the Radwaste Building based on distance and direction to the Control Room receptors, and the PCCS vent χ/Q values are used to evaluate releases from the Radwaste Building in DCD Section 15.3.16. The PCCS χ/Q values are compared to the Radwaste Building χ/Q results.
Note 2: These results are for confirmation of the Radwaste Building to Control Room Unfiltered Inleakage χ/Q values.
Note 3: These results are for confirmation of the Radwaste Building to Control Room Intake χ/Q values.

NAPS ESP COL 2.3-2

Table 2.3-206 Unit 3 Blowout Panels/Reactor Building χ/Q Results (sec/m^3)

Source/Receptor	0–2 hr	2–8 hr	8–24 hr	1–4 d	4–30 d
BPN to CBL[1]	2.04E-03	1.67E-03	7.21E-04	4.93E-04	4.20E-04
BPS to CBL[1]	2.16E-03	1.72E-03	6.72E-04	5.25E-04	3.94E-04
BPN to EN[2]	1.78E-03	1.30E-03	5.04E-04	3.72E-04	3.01E-04
BPN to ES[2]	1.43E-03	1.13E-03	4.89E-04	3.44E-04	2.86E-04
BPN to N[2]	1.41E-03	1.15E-03	4.96E-04	3.40E-04	2.93E-04
BPS to EN[2]	1.52E-03	1.16E-03	4.22E-04	3.27E-04	2.64E-04
BPS to ES[2]	1.78E-03	1.25E-03	4.63E-04	3.35E-04	2.73E-04
BPS to N[2]	2.00E-03	1.38E-03	5.23E-04	3.66E-04	3.06E-04

Note 1: These results are for confirmation of the Reactor Building Blowout Panels to Control Room Unfiltered Inleakage χ/Q values.

Note 2: These results are for confirmation of the Reactor Building Blowout Panels to Control Room Intake χ/Q values.

NAPS ESP COL 2.3-2

Table 2.3-207 Unit 3 Cross Unit χ/Q Results (sec/m^3)

Source/Receptor	0–2 hr	2–8 hr	8–24 hr	1–4 d	4–30 d
Unit 1/2 Release to Unit 3	5.13E-05	3.67E-05	1.36E-05	9.95E-06	7.51E-06

NAPS COL 2.0-11-A

Table 2.3-208 Long-Term χ/Q (sec/m^3) for Routine Releases at Distances Between 0.25 to 50 Miles, No Decay, Undepleted

Ground Level Release - No Purge Releases

Sector	Distance in Miles from the Site										
	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	8.349E-06	2.976E-06	1.595E-06	1.023E-06	5.508E-07	3.558E-07	2.538E-07	1.928E-07	1.529E-07	1.252E-07	1.050E-07
SSW	6.537E-06	2.338E-06	1.261E-06	8.122E-07	4.388E-07	2.841E-07	2.030E-07	1.544E-07	1.226E-07	1.005E-07	8.434E-08
SW	5.863E-06	2.085E-06	1.125E-06	7.259E-07	3.931E-07	2.550E-07	1.825E-07	1.390E-07	1.105E-07	9.067E-08	7.617E-08
WSW	5.511E-06	1.940E-06	1.044E-06	6.739E-07	3.656E-07	2.375E-07	1.702E-07	1.298E-07	1.033E-07	8.482E-08	7.132E-08
W	6.877E-06	2.365E-06	1.265E-06	8.167E-07	4.457E-07	2.913E-07	2.098E-07	1.606E-07	1.282E-07	1.056E-07	8.904E-08
WNW	6.006E-06	2.046E-06	1.097E-06	7.084E-07	3.860E-07	2.519E-07	1.812E-07	1.387E-07	1.107E-07	9.113E-08	7.682E-08
NW	6.009E-06	2.064E-06	1.122E-06	7.288E-07	4.001E-07	2.624E-07	1.895E-07	1.454E-07	1.163E-07	9.597E-08	8.104E-08
NNW	5.110E-06	1.747E-06	9.583E-07	6.266E-07	3.458E-07	2.274E-07	1.645E-07	1.264E-07	1.013E-07	8.362E-08	7.067E-08
N	1.299E-05	4.468E-06	2.462E-06	1.613E-06	8.890E-07	5.834E-07	4.214E-07	3.234E-07	2.588E-07	2.136E-07	1.803E-07
NNE	1.657E-05	5.654E-06	3.098E-06	2.029E-06	1.119E-06	7.350E-07	5.312E-07	4.079E-07	3.265E-07	2.695E-07	2.276E-07
NE	1.352E-05	4.622E-06	2.530E-06	1.656E-06	9.142E-07	6.013E-07	4.350E-07	3.343E-07	2.679E-07	2.212E-07	1.870E-07
ENE	8.502E-06	2.817E-06	1.532E-06	1.007E-06	5.622E-07	3.730E-07	2.717E-07	2.100E-07	1.690E-07	1.401E-07	1.188E-07
E	1.668E-05	5.305E-06	2.852E-06	1.885E-06	1.069E-06	7.183E-07	5.283E-07	4.114E-07	3.333E-07	2.779E-07	2.368E-07
ESE	2.566E-05	7.927E-06	4.114E-06	2.670E-06	1.524E-06	1.038E-06	7.709E-07	6.052E-07	4.936E-07	4.140E-07	3.546E-07
SE	1.818E-05	5.672E-06	2.914E-06	1.868E-06	1.056E-06	7.154E-07	5.298E-07	4.149E-07	3.378E-07	2.828E-07	2.420E-07
SSE	9.287E-06	3.113E-06	1.640E-06	1.051E-06	5.752E-07	3.782E-07	2.737E-07	2.104E-07	1.687E-07	1.394E-07	1.179E-07

NAPS COL 2.0-11-A

Table 2.3-208 Long-Term χ/Q (sec/m^3) for Routine Releases at Distances Between 0.25 to 50 Miles, No Decay, Undepleted

Ground Level Release - No Purge Releases

	Distance in Miles from the Site										
Sector	**5.000**	**7.500**	**10.000**	**15.000**	**20.000**	**25.000**	**30.000**	**35.000**	**40.000**	**45.000**	**50.000**
S	8.977E-08	4.929E-08	3.232E-08	1.794E-08	1.188E-08	8.646E-09	6.678E-09	5.373E-09	4.453E-09	3.776E-09	3.259E-09
SSW	7.215E-08	3.970E-08	2.608E-08	1.450E-08	9.599E-09	6.984E-09	5.393E-09	4.338E-09	3.595E-09	3.047E-09	2.629E-09
SW	6.521E-08	3.601E-08	2.372E-08	1.324E-08	8.788E-09	6.409E-09	4.959E-09	3.995E-09	3.315E-09	2.813E-09	2.430E-09
WSW	6.111E-08	3.386E-08	2.236E-08	1.253E-08	8.344E-09	6.101E-09	4.730E-09	3.818E-09	3.174E-09	2.697E-09	2.333E-09
W	7.648E-08	4.280E-08	2.847E-08	1.613E-08	1.083E-08	7.971E-09	6.213E-09	5.038E-09	4.205E-09	3.587E-09	3.113E-09
WNW	6.599E-08	3.696E-08	2.460E-08	1.396E-08	9.406E-09	6.937E-09	5.417E-09	4.399E-09	3.676E-09	3.139E-09	2.727E-09
NW	6.970E-08	3.920E-08	2.616E-08	1.488E-08	1.002E-08	7.391E-09	5.770E-09	4.684E-09	3.913E-09	3.340E-09	2.900E-09
NNW	6.083E-08	3.431E-08	2.294E-08	1.307E-08	8.809E-09	6.497E-09	5.072E-09	4.118E-09	3.439E-09	2.935E-09	2.548E-09
N	1.551E-07	8.723E-08	5.819E-08	3.307E-08	2.223E-08	1.637E-08	1.276E-08	1.034E-08	8.630E-09	7.358E-09	6.382E-09
NNE	1.958E-07	1.103E-07	7.363E-08	4.190E-08	2.821E-08	2.079E-08	1.622E-08	1.316E-08	1.099E-08	9.374E-09	8.135E-09
NE	1.609E-07	9.075E-08	6.066E-08	3.457E-08	2.329E-08	1.718E-08	1.341E-08	1.089E-08	9.095E-09	7.763E-09	6.739E-09
ENE	1.026E-07	5.856E-08	3.948E-08	2.277E-08	1.547E-08	1.148E-08	9.008E-09	7.345E-09	6.158E-09	5.273E-09	4.592E-09
E	2.053E-07	1.190E-07	8.114E-08	4.750E-08	3.260E-08	2.439E-08	1.926E-08	1.579E-08	1.330E-08	1.144E-08	9.993E-09
ESE	3.089E-07	1.823E-07	1.258E-07	7.493E-08	5.206E-08	3.932E-08	3.130E-08	2.583E-08	2.188E-08	1.891E-08	1.660E-08
SE	2.106E-07	1.239E-07	8.534E-08	5.075E-08	3.524E-08	2.661E-08	2.118E-08	1.748E-08	1.481E-08	1.280E-08	1.124E-08
SSE	1.016E-07	5.751E-08	3.860E-08	2.216E-08	1.504E-08	1.116E-08	8.765E-09	7.150E-09	5.999E-09	5.141E-09	4.480E-09

NAPS COL 2.0-11-A

Table 2.3-209 Long-Term χ/Q (sec/m^3) for Routine Releases Along Various Distance Segments, No Decay, Undepleted

Ground Level Release - No Purge Releases

Direction From Site	Segment Boundaries in Miles from the Site									
	0.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	1.648E-06	5.691E-07	2.566E-07	1.538E-07	1.054E-07	5.074E-08	1.844E-08	8.721E-09	5.395E-09	3.785E-09
SSW	1.301E-06	4.530E-07	2.052E-07	1.233E-07	8.461E-08	4.086E-08	1.489E-08	7.045E-09	4.357E-09	3.055E-09
SW	1.161E-06	4.057E-07	1.845E-07	1.111E-07	7.641E-08	3.704E-08	1.359E-08	6.463E-09	4.011E-09	2.820E-09
WSW	1.079E-06	3.772E-07	1.720E-07	1.038E-07	7.154E-08	3.480E-08	1.285E-08	6.151E-09	3.833E-09	2.704E-09
W	1.310E-06	4.595E-07	2.118E-07	1.289E-07	8.930E-08	4.392E-08	1.652E-08	8.030E-09	5.056E-09	3.594E-09
WNW	1.135E-06	3.980E-07	1.830E-07	1.112E-07	7.705E-08	3.792E-08	1.430E-08	6.988E-09	4.415E-09	3.146E-09
NW	1.157E-06	4.120E-07	1.913E-07	1.169E-07	8.126E-08	4.018E-08	1.523E-08	7.444E-09	4.700E-09	3.347E-09
NNW	9.862E-07	3.556E-07	1.660E-07	1.017E-07	7.086E-08	3.515E-08	1.337E-08	6.544E-09	4.132E-09	2.941E-09
N	2.530E-06	9.140E-07	4.254E-07	2.601E-07	1.808E-07	8.941E-08	3.383E-08	1.649E-08	1.038E-08	7.373E-09
NNE	3.191E-06	1.151E-06	5.362E-07	3.280E-07	2.283E-07	1.130E-07	4.287E-08	2.094E-08	1.321E-08	9.393E-09
NE	2.606E-06	9.399E-07	4.391E-07	2.691E-07	1.875E-07	9.297E-08	3.536E-08	1.730E-08	1.093E-08	7.778E-09
ENE	1.584E-06	5.770E-07	2.740E-07	1.697E-07	1.191E-07	5.987E-08	2.324E-08	1.155E-08	7.368E-09	5.283E-09
E	2.967E-06	1.094E-06	5.322E-07	3.345E-07	2.373E-07	1.214E-07	4.835E-08	2.453E-08	1.583E-08	1.145E-08
ESE	4.319E-06	1.563E-06	7.757E-07	4.952E-07	3.553E-07	1.853E-07	7.606E-08	3.951E-08	2.588E-08	1.893E-08
SE	3.062E-06	1.085E-06	5.334E-07	3.389E-07	2.425E-07	1.260E-07	5.154E-08	2.674E-08	1.752E-08	1.282E-08
SSE	1.705E-06	5.933E-07	2.763E-07	1.695E-07	1.182E-07	5.889E-08	2.265E-08	1.124E-08	7.173E-09	5.150E-09

NAPS COL 2.0-11-A

Table 2.3-210 Long-Term χ/Q (sec/m^3) for Routine Releases at Distances Between 0.25 to 50 Miles, 2.260 Day Decay, Undepleted

Ground Level Release - No Purge Releases

	Distance in Miles from the Site										
Sector	**0.250**	**0.500**	**0.750**	**1.000**	**1.500**	**2.000**	**2.500**	**3.000**	**3.500**	**4.000**	**4.500**
S	8.340E-06	2.969E-06	1.590E-06	1.019E-06	5.474E-07	3.529E-07	2.512E-07	1.904E-07	1.507E-07	1.231E-07	1.030E-07
SSW	6.530E-06	2.333E-06	1.257E-06	8.086E-07	4.359E-07	2.816E-07	2.007E-07	1.523E-07	1.207E-07	9.866E-08	8.262E-08
SW	5.856E-06	2.080E-06	1.121E-06	7.224E-07	3.903E-07	2.526E-07	1.804E-07	1.370E-07	1.087E-07	8.892E-08	7.452E-08
WSW	5.504E-06	1.936E-06	1.041E-06	6.705E-07	3.628E-07	2.351E-07	1.681E-07	1.278E-07	1.015E-07	8.308E-08	6.967E-08
W	6.868E-06	2.359E-06	1.260E-06	8.125E-07	4.423E-07	2.883E-07	2.070E-07	1.581E-07	1.259E-07	1.034E-07	8.693E-08
WNW	5.998E-06	2.041E-06	1.093E-06	7.049E-07	3.831E-07	2.494E-07	1.789E-07	1.366E-07	1.087E-07	8.928E-08	7.507E-08
NW	6.001E-06	2.059E-06	1.117E-06	7.252E-07	3.971E-07	2.598E-07	1.871E-07	1.432E-07	1.143E-07	9.404E-08	7.920E-08
NNW	5.103E-06	1.742E-06	9.543E-07	6.231E-07	3.429E-07	2.248E-07	1.622E-07	1.243E-07	9.926E-08	8.173E-08	6.888E-08
N	1.297E-05	4.455E-06	2.452E-06	1.604E-06	8.816E-07	5.770E-07	4.156E-07	3.181E-07	2.538E-07	2.088E-07	1.759E-07
NNE	1.655E-05	5.639E-06	3.086E-06	2.019E-06	1.110E-06	7.273E-07	5.242E-07	4.014E-07	3.205E-07	2.638E-07	2.222E-07
NE	1.350E-05	4.610E-06	2.520E-06	1.647E-06	9.071E-07	5.950E-07	4.294E-07	3.291E-07	2.630E-07	2.166E-07	1.826E-07
ENE	8.490E-06	2.809E-06	1.525E-06	1.001E-06	5.574E-07	3.687E-07	2.678E-07	2.063E-07	1.656E-07	1.369E-07	1.158E-07
E	1.665E-05	5.288E-06	2.839E-06	1.874E-06	1.059E-06	7.094E-07	5.201E-07	4.038E-07	3.261E-07	2.710E-07	2.302E-07
ESE	2.562E-05	7.901E-06	4.094E-06	2.653E-06	1.509E-06	1.024E-06	7.584E-07	5.935E-07	4.825E-07	4.033E-07	3.443E-07
SE	1.815E-05	5.654E-06	2.900E-06	1.857E-06	1.046E-06	7.064E-07	5.213E-07	4.070E-07	3.302E-07	2.756E-07	2.350E-07
SSE	9.275E-06	3.105E-06	1.634E-06	1.045E-06	5.708E-07	3.743E-07	2.701E-07	2.071E-07	1.656E-07	1.364E-07	1.151E-07

NAPS COL 2.0-11-A

Table 2.3-210 Long-Term χ/Q (sec/m^3) for Routine Releases at Distances Between 0.25 to 50 Miles, 2.260 Day Decay, Undepleted

Ground Level Release - No Purge Releases

	Distance in Miles from the Site										
Sector	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	8.787E-08	4.771E-08	3.094E-08	1.680E-08	1.087E-08	7.736E-09	5.842E-09	4.596E-09	3.725E-09	3.089E-09	2.607E-09
SSW	7.050E-08	3.834E-08	2.489E-08	1.351E-08	8.731E-09	6.203E-09	4.677E-09	3.673E-09	2.972E-09	2.460E-09	2.074E-09
SW	6.364E-08	3.471E-08	2.257E-08	1.228E-08	7.951E-09	5.654E-09	4.265E-09	3.351E-09	2.712E-09	2.244E-09	1.891E-09
WSW	5.954E-08	3.256E-08	2.121E-08	1.157E-08	7.502E-09	5.340E-09	4.031E-09	3.168E-09	2.564E-09	2.123E-09	1.788E-09
W	7.446E-08	4.111E-08	2.697E-08	1.486E-08	9.706E-09	6.949E-09	5.269E-09	4.157E-09	3.376E-09	2.802E-09	2.367E-09
WNW	6.431E-08	3.555E-08	2.335E-08	1.291E-08	8.466E-09	6.082E-09	4.626E-09	3.660E-09	2.980E-09	2.479E-09	2.099E-09
NW	6.795E-08	3.772E-08	2.484E-08	1.377E-08	9.036E-09	6.493E-09	4.940E-09	3.908E-09	3.182E-09	2.648E-09	2.242E-09
NNW	5.912E-08	3.287E-08	2.166E-08	1.200E-08	7.858E-09	5.634E-09	4.276E-09	3.375E-09	2.741E-09	2.276E-09	1.922E-09
N	1.508E-07	8.364E-08	5.502E-08	3.040E-08	1.988E-08	1.424E-08	1.080E-08	8.516E-09	6.914E-09	5.737E-09	4.844E-09
NNE	1.907E-07	1.059E-07	6.976E-08	3.863E-08	2.532E-08	1.816E-08	1.380E-08	1.090E-08	8.864E-09	7.367E-09	6.228E-09
NE	1.567E-07	8.721E-08	5.752E-08	3.192E-08	2.094E-08	1.504E-08	1.144E-08	9.046E-09	7.361E-09	6.123E-09	5.181E-09
ENE	9.965E-08	5.604E-08	3.722E-08	2.084E-08	1.375E-08	9.910E-09	7.553E-09	5.983E-09	4.873E-09	4.055E-09	3.432E-09
E	1.990E-07	1.136E-07	7.620E-08	4.324E-08	2.877E-08	2.087E-08	1.598E-08	1.271E-08	1.038E-08	8.662E-09	7.346E-09
ESE	2.989E-07	1.735E-07	1.178E-07	6.789E-08	4.566E-08	3.339E-08	2.573E-08	2.057E-08	1.688E-08	1.413E-08	1.202E-08
SE	2.038E-07	1.179E-07	7.991E-08	4.598E-08	3.091E-08	2.259E-08	1.741E-08	1.391E-08	1.142E-08	9.560E-09	8.134E-09
SSE	9.884E-08	5.519E-08	3.652E-08	2.038E-08	1.344E-08	9.697E-09	7.400E-09	5.869E-09	4.787E-09	3.989E-09	3.381E-09

NAPS COL 2.0-11-A

Table 2.3-211 Long-Term χ/Q (sec/m^3) for Routine Releases Along Various Distance Segments, 2.260 Day Decay, Undepleted

Direction From Site	Ground Level Release - No Purge Releases: Segment Boundaries in Miles from the Site									
	0.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	1.643E-06	5.658E-07	2.540E-07	1.515E-07	1.034E-07	4.918E-08	1.731E-08	7.815E-09	4.620E-09	3.099E-09
SSW	1.297E-06	4.501E-07	2.029E-07	1.213E-07	8.288E-08	3.951E-08	1.391E-08	6.267E-09	3.693E-09	2.469E-09
SW	1.157E-06	4.029E-07	1.823E-07	1.092E-07	7.476E-08	3.574E-08	1.264E-08	5.711E-09	3.368E-09	2.252E-09
WSW	1.075E-06	3.744E-07	1.699E-07	1.020E-07	6.989E-08	3.351E-08	1.190E-08	5.393E-09	3.185E-09	2.130E-09
W	1.305E-06	4.561E-07	2.091E-07	1.265E-07	8.719E-08	4.224E-08	1.526E-08	7.012E-09	4.177E-09	2.811E-09
WNW	1.131E-06	3.952E-07	1.808E-07	1.093E-07	7.530E-08	3.652E-08	1.325E-08	6.135E-09	3.677E-09	2.487E-09
NW	1.152E-06	4.090E-07	1.889E-07	1.148E-07	7.943E-08	3.871E-08	1.413E-08	6.550E-09	3.926E-09	2.656E-09
NNW	9.822E-07	3.527E-07	1.637E-07	9.973E-08	6.907E-08	3.372E-08	1.231E-08	5.684E-09	3.391E-09	2.283E-09
N	2.520E-06	9.067E-07	4.196E-07	2.551E-07	1.764E-07	8.585E-08	3.120E-08	1.437E-08	8.557E-09	5.755E-09
NNE	3.179E-06	1.142E-06	5.292E-07	3.220E-07	2.228E-07	1.087E-07	3.963E-08	1.832E-08	1.095E-08	7.389E-09
NE	2.597E-06	9.328E-07	4.335E-07	2.642E-07	1.831E-07	8.946E-08	3.273E-08	1.517E-08	9.088E-09	6.141E-09
ENE	1.578E-06	5.722E-07	2.701E-07	1.663E-07	1.160E-07	5.737E-08	2.133E-08	9.991E-09	6.009E-09	4.067E-09
E	2.954E-06	1.085E-06	5.241E-07	3.273E-07	2.307E-07	1.159E-07	4.413E-08	2.102E-08	1.276E-08	8.685E-09
ESE	4.300E-06	1.548E-06	7.634E-07	4.840E-07	3.450E-07	1.766E-07	6.909E-08	3.360E-08	2.064E-08	1.416E-08
SE	3.048E-06	1.075E-06	5.249E-07	3.313E-07	2.355E-07	1.201E-07	4.682E-08	2.274E-08	1.396E-08	9.582E-09
SSE	1.699E-06	5.889E-07	2.727E-07	1.663E-07	1.154E-07	5.659E-08	2.088E-08	9.777E-09	5.894E-09	4.001E-09

NAPS COL 2.0-11-A

Table 2.3-212 Long-Term χ/Q (sec/m^3) for Routine Releases at Distances Between 0.25 to 50 Miles, 8.000 Day Decay, Depleted

Ground Level Release - No Purge Releases

	Distance in Miles from the Site										
Sector	**0.250**	**0.500**	**0.750**	**1.000**	**1.500**	**2.000**	**2.500**	**3.000**	**3.500**	**4.000**	**4.500**
S	7.899E-06	2.716E-06	1.420E-06	8.947E-07	4.669E-07	2.939E-07	2.050E-07	1.526E-07	1.188E-07	9.566E-08	7.897E-08
SSW	6.185E-06	2.134E-06	1.122E-06	7.101E-07	3.720E-07	2.347E-07	1.639E-07	1.222E-07	9.526E-08	7.674E-08	6.340E-08
SW	5.547E-06	1.902E-06	1.002E-06	6.345E-07	3.332E-07	2.106E-07	1.474E-07	1.100E-07	8.583E-08	6.922E-08	5.723E-08
WSW	5.214E-06	1.771E-06	9.297E-07	5.891E-07	3.098E-07	1.961E-07	1.374E-07	1.027E-07	8.020E-08	6.473E-08	5.357E-08
W	6.506E-06	2.158E-06	1.126E-06	7.138E-07	3.777E-07	2.405E-07	1.693E-07	1.270E-07	9.954E-08	8.058E-08	6.686E-08
WNW	5.682E-06	1.867E-06	9.770E-07	6.193E-07	3.271E-07	2.080E-07	1.463E-07	1.097E-07	8.593E-08	6.955E-08	5.770E-08
NW	5.685E-06	1.884E-06	9.984E-07	6.371E-07	3.391E-07	2.167E-07	1.529E-07	1.150E-07	9.032E-08	7.325E-08	6.088E-08
NNW	4.835E-06	1.594E-06	8.530E-07	5.476E-07	2.930E-07	1.877E-07	1.327E-07	9.991E-08	7.856E-08	6.378E-08	5.304E-08
N	1.229E-05	4.077E-06	2.192E-06	1.410E-06	7.532E-07	4.816E-07	3.400E-07	2.557E-07	2.009E-07	1.629E-07	1.354E-07
NNE	1.568E-05	5.159E-06	2.758E-06	1.774E-06	9.485E-07	6.068E-07	4.287E-07	3.225E-07	2.534E-07	2.056E-07	1.709E-07
NE	1.279E-05	4.218E-06	2.252E-06	1.447E-06	7.747E-07	4.964E-07	3.511E-07	2.644E-07	2.079E-07	1.688E-07	1.404E-07
ENE	8.043E-06	2.570E-06	1.363E-06	8.802E-07	4.763E-07	3.079E-07	2.192E-07	1.660E-07	1.311E-07	1.068E-07	8.918E-08
E	1.578E-05	4.840E-06	2.539E-06	1.647E-06	9.054E-07	5.927E-07	4.260E-07	3.251E-07	2.584E-07	2.118E-07	1.776E-07
ESE	2.428E-05	7.232E-06	3.661E-06	2.333E-06	1.291E-06	8.561E-07	6.216E-07	4.781E-07	3.827E-07	3.154E-07	2.659E-07
SE	1.720E-05	5.175E-06	2.593E-06	1.633E-06	8.942E-07	5.903E-07	4.272E-07	3.278E-07	2.619E-07	2.155E-07	1.814E-07
SSE	8.786E-06	2.841E-06	1.460E-06	9.185E-07	4.874E-07	3.122E-07	2.209E-07	1.664E-07	1.309E-07	1.064E-07	8.852E-08

NAPS COL 2.0-11-A

Table 2.3-212 Long-Term χ/Q (sec/m^3) for Routine Releases at Distances Between 0.25 to 50 Miles, 8.000 Day Decay, Depleted

Ground Level Release - No Purge Releases

	Distance in Miles from the Site										
Sector	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	6.651E-08	3.443E-08	2.145E-08	1.095E-08	6.764E-09	4.634E-09	3.389E-09	2.593E-09	2.050E-09	1.663E-09	1.376E-09
SSW	5.343E-08	2.771E-08	1.729E-08	8.835E-09	5.456E-09	3.735E-09	2.730E-09	2.087E-09	1.650E-09	1.337E-09	1.106E-09
SW	4.828E-08	2.512E-08	1.571E-08	8.057E-09	4.988E-09	3.421E-09	2.504E-09	1.917E-09	1.517E-09	1.230E-09	1.018E-09
WSW	4.522E-08	2.361E-08	1.480E-08	7.614E-09	4.727E-09	3.249E-09	2.383E-09	1.827E-09	1.447E-09	1.175E-09	9.732E-10
W	5.658E-08	2.983E-08	1.883E-08	9.796E-09	6.130E-09	4.240E-09	3.125E-09	2.406E-09	1.913E-09	1.559E-09	1.295E-09
WNW	4.883E-08	2.577E-08	1.629E-08	8.491E-09	5.330E-09	3.696E-09	2.730E-09	2.106E-09	1.677E-09	1.369E-09	1.139E-09
NW	5.158E-08	2.733E-08	1.732E-08	9.051E-09	5.682E-09	3.940E-09	2.910E-09	2.244E-09	1.787E-09	1.458E-09	1.212E-09
NNW	4.498E-08	2.389E-08	1.516E-08	7.933E-09	4.979E-09	3.451E-09	2.547E-09	1.963E-09	1.562E-09	1.274E-09	1.058E-09
N	1.147E-07	6.077E-08	3.848E-08	2.008E-08	1.258E-08	8.703E-09	6.415E-09	4.939E-09	3.926E-09	3.198E-09	2.655E-09
NNE	1.449E-07	7.685E-08	4.871E-08	2.546E-08	1.597E-08	1.107E-08	8.167E-09	6.294E-09	5.008E-09	4.082E-09	3.393E-09
NE	1.191E-07	6.325E-08	4.014E-08	2.101E-08	1.320E-08	9.151E-09	6.758E-09	5.211E-09	4.149E-09	3.384E-09	2.813E-09
ENE	7.585E-08	4.077E-08	2.608E-08	1.381E-08	8.733E-09	6.090E-09	4.516E-09	3.495E-09	2.791E-09	2.282E-09	1.901E-09
E	1.517E-07	8.281E-08	5.355E-08	2.876E-08	1.837E-08	1.291E-08	9.628E-09	7.488E-09	6.004E-09	4.927E-09	4.118E-09
ESE	2.281E-07	1.267E-07	8.293E-08	4.530E-08	2.928E-08	2.076E-08	1.560E-08	1.221E-08	9.839E-09	8.111E-09	6.808E-09
SE	1.555E-07	8.612E-08	5.627E-08	3.068E-08	1.982E-08	1.405E-08	1.056E-08	8.261E-09	6.659E-09	5.490E-09	4.608E-09
SSE	7.512E-08	4.007E-08	2.552E-08	1.345E-08	8.506E-09	5.932E-09	4.402E-09	3.409E-09	2.724E-09	2.229E-09	1.859E-09

NAPS COL 2.0-11-A

Table 2.3-213 Long-Term χ/Q (sec/m^3) for Routine Releases Along Various Distance Segments, 8.000 Day Decay, Depleted

Ground Level Release - No Purge Releases

Segment Boundaries in Miles from the Site

Direction From Site	0.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	1.474E-06	4.851E-07	2.078E-07	1.197E-07	7.930E-08	3.579E-08	1.142E-08	4.704E-09	2.613E-09	1.671E-09
SSW	1.164E-06	3.861E-07	1.661E-07	9.590E-08	6.366E-08	2.879E-08	9.212E-09	3.792E-09	2.104E-09	1.344E-09
SW	1.039E-06	3.457E-07	1.493E-07	8.640E-08	5.747E-08	2.608E-08	8.394E-09	3.472E-09	1.932E-09	1.237E-09
WSW	9.652E-07	3.213E-07	1.392E-07	8.073E-08	5.378E-08	2.449E-08	7.927E-09	3.297E-09	1.841E-09	1.181E-09
W	1.172E-06	3.914E-07	1.714E-07	1.002E-07	6.712E-08	3.089E-08	1.018E-08	4.298E-09	2.424E-09	1.566E-09
WNW	1.016E-06	3.391E-07	1.481E-07	8.647E-08	5.793E-08	2.668E-08	8.818E-09	3.746E-09	2.121E-09	1.375E-09
NW	1.035E-06	3.509E-07	1.548E-07	9.087E-08	6.110E-08	2.827E-08	9.391E-09	3.993E-09	2.260E-09	1.465E-09
NNW	8.820E-07	3.028E-07	1.342E-07	7.903E-08	5.324E-08	2.470E-08	8.226E-09	3.497E-09	1.977E-09	1.279E-09
N	2.263E-06	7.783E-07	3.440E-07	2.021E-07	1.359E-07	6.285E-08	2.083E-08	8.820E-09	4.975E-09	3.213E-09
NNE	2.854E-06	9.800E-07	4.337E-07	2.550E-07	1.716E-07	7.946E-08	2.641E-08	1.122E-08	6.339E-09	4.101E-09
NE	2.331E-06	8.004E-07	3.552E-07	2.092E-07	1.409E-07	6.538E-08	2.179E-08	9.272E-09	5.248E-09	3.399E-09
ENE	1.417E-06	4.912E-07	2.215E-07	1.318E-07	8.948E-08	4.204E-08	1.428E-08	6.165E-09	3.519E-09	2.292E-09
E	2.654E-06	9.313E-07	4.301E-07	2.597E-07	1.781E-07	8.511E-08	2.965E-08	1.305E-08	7.534E-09	4.946E-09
ESE	3.864E-06	1.329E-06	6.267E-07	3.843E-07	2.666E-07	1.298E-07	4.654E-08	2.097E-08	1.227E-08	8.140E-09
SE	2.740E-06	9.232E-07	4.309E-07	2.631E-07	1.819E-07	8.828E-08	3.154E-08	1.419E-08	8.307E-09	5.510E-09
SSE	1.526E-06	5.054E-07	2.235E-07	1.317E-07	8.884E-08	4.140E-08	1.394E-08	6.007E-09	3.432E-09	2.239E-09

NAPS COL 2.0-11-A

Table 2.3-214 Long-Term D/Q ($1/m^2$) for Routine Releases at Distances Between 0.25 to 50 Miles

Ground Level Release - No Purge Releases
Relative Deposition Per Unit Area ($1/m^2$) At Fixed Points By Downwind Sectors
Distances In Miles

Direction From Site	0.25	0.50	0.75	1.00	1.50	2.00	2.50	3.00	3.50	4.00	4.50
S	4.819E-08	1.630E-08	8.367E-09	5.138E-09	2.561E-09	1.553E-09	1.050E-09	7.611E-10	5.787E-10	4.559E-10	3.691E-10
SSW	3.194E-08	1.080E-08	5.546E-09	3.405E-09	1.698E-09	1.030E-09	6.961E-10	5.045E-10	3.836E-10	3.022E-10	2.446E-10
SW	2.633E-08	8.902E-09	4.571E-09	2.807E-09	1.399E-09	8.486E-10	5.738E-10	4.158E-10	3.161E-10	2.491E-10	2.016E-10
WSW	2.286E-08	7.732E-09	3.970E-09	2.438E-09	1.215E-09	7.371E-10	4.983E-10	3.611E-10	2.746E-10	2.163E-10	1.751E-10
W	2.691E-08	9.101E-09	4.673E-09	2.869E-09	1.430E-09	8.676E-10	5.866E-10	4.251E-10	3.232E-10	2.546E-10	2.061E-10
WNW	2.495E-08	8.438E-09	4.333E-09	2.660E-09	1.326E-09	8.044E-10	5.439E-10	3.941E-10	2.997E-10	2.361E-10	1.911E-10
NW	2.242E-08	7.583E-09	3.893E-09	2.391E-09	1.192E-09	7.229E-10	4.887E-10	3.542E-10	2.693E-10	2.122E-10	1.718E-10
NNW	1.628E-08	5.504E-09	2.826E-09	1.735E-09	8.652E-10	5.247E-10	3.548E-10	2.571E-10	1.955E-10	1.540E-10	1.247E-10
N	4.309E-08	1.457E-08	7.481E-09	4.594E-09	2.290E-09	1.389E-09	9.391E-10	6.805E-10	5.175E-10	4.077E-10	3.300E-10
NNE	6.257E-08	2.116E-08	1.086E-08	6.671E-09	3.326E-09	2.017E-09	1.364E-09	9.882E-10	7.514E-10	5.920E-10	4.793E-10
NE	5.046E-08	1.706E-08	8.761E-09	5.379E-09	2.682E-09	1.627E-09	1.100E-09	7.969E-10	6.059E-10	4.774E-10	3.865E-10
ENE	2.720E-08	9.199E-09	4.723E-09	2.900E-09	1.446E-09	8.769E-10	5.929E-10	4.296E-10	3.267E-10	2.574E-10	2.084E-10
E	3.824E-08	1.293E-08	6.640E-09	4.077E-09	2.033E-09	1.233E-09	8.335E-10	6.040E-10	4.593E-10	3.618E-10	2.929E-10
ESE	5.097E-08	1.724E-08	8.849E-09	5.434E-09	2.709E-09	1.643E-09	1.111E-09	8.050E-10	6.121E-10	4.822E-10	3.904E-10
SE	4.574E-08	1.547E-08	7.942E-09	4.877E-09	2.431E-09	1.475E-09	9.970E-10	7.225E-10	5.493E-10	4.328E-10	3.504E-10
SSE	4.085E-08	1.381E-08	7.092E-09	4.355E-09	2.171E-09	1.317E-09	8.902E-10	6.451E-10	4.905E-10	3.865E-10	3.129E-10

NAPS COL 2.0-11-A

Table 2.3-214 Long-Term D/Q ($1/m^2$) for Routine Releases at Distances Between 0.25 to 50 Miles

Ground Level Release - No Purge Releases
Relative Deposition Per Unit Area ($1/m^2$) At Fixed Points By Downwind Sectors
Distances In Miles

Direction From Site	5.00	7.50	10.00	15.00	20.00	25.00	30.00	35.00	40.00	45.00	50.00
S	3.053E-10	1.496E-10	9.388E-11	4.745E-11	2.872E-11	1.926E-11	1.380E-11	1.036E-11	8.056E-12	6.435E-12	5.252E-12
SSW	2.024E-10	9.917E-11	6.222E-11	3.145E-11	1.904E-11	1.276E-11	9.145E-12	6.867E-12	5.339E-12	4.265E-12	3.481E-12
SW	1.668E-10	8.174E-11	5.129E-11	2.592E-11	1.569E-11	1.052E-11	7.538E-12	5.660E-12	4.401E-12	3.515E-12	2.869E-12
WSW	1.449E-10	7.099E-11	4.454E-11	2.251E-11	1.363E-11	9.136E-12	6.547E-12	4.916E-12	3.822E-12	3.053E-12	2.492E-12
W	1.705E-10	8.356E-11	5.243E-11	2.650E-11	1.604E-11	1.075E-11	7.706E-12	5.786E-12	4.499E-12	3.594E-12	2.933E-12
WNW	1.581E-10	7.748E-11	4.861E-11	2.457E-11	1.487E-11	9.971E-12	7.145E-12	5.365E-12	4.171E-12	3.332E-12	2.720E-12
NW	1.421E-10	6.962E-11	4.369E-11	2.208E-11	1.336E-11	8.961E-12	6.421E-12	4.821E-12	3.749E-12	2.994E-12	2.444E-12
NNW	1.031E-10	5.054E-11	3.171E-11	1.603E-11	9.701E-12	6.504E-12	4.661E-12	3.500E-12	2.721E-12	2.174E-12	1.774E-12
N	2.730E-10	1.338E-10	8.394E-11	4.243E-11	2.568E-11	1.722E-11	1.234E-11	9.264E-12	7.203E-12	5.754E-12	4.697E-12
NNE	3.964E-10	1.943E-10	1.219E-10	6.161E-11	3.729E-11	2.500E-11	1.792E-11	1.345E-11	1.046E-11	8.355E-12	6.820E-12
NE	3.197E-10	1.567E-10	9.830E-11	4.968E-11	3.007E-11	2.016E-11	1.445E-11	1.085E-11	8.435E-12	6.738E-12	5.500E-12
ENE	1.724E-10	8.446E-11	5.300E-11	2.679E-11	1.621E-11	1.087E-11	7.789E-12	5.849E-12	4.548E-12	3.633E-12	2.965E-12
E	2.423E-10	1.187E-10	7.451E-11	3.766E-11	2.279E-11	1.528E-11	1.095E-11	8.223E-12	6.393E-12	5.107E-12	4.168E-12
ESE	3.229E-10	1.583E-10	9.929E-11	5.019E-11	3.038E-11	2.037E-11	1.459E-11	1.096E-11	8.520E-12	6.806E-12	5.555E-12
SE	2.898E-10	1.420E-10	8.912E-11	4.504E-11	2.726E-11	1.828E-11	1.310E-11	9.835E-12	7.647E-12	6.108E-12	4.986E-12
SSE	2.588E-10	1.268E-10	7.957E-11	4.022E-11	2.434E-11	1.632E-11	1.170E-11	8.782E-12	6.828E-12	5.454E-12	4.452E-12

NAPS COL 2.0-11-A

Table 2.3-215 Long-Term D/Q ($1/m^2$) for Routine Releases Along Various Distance Segments

Ground Level Release - No Purge Release
Relative Deposition Per Unit Area ($1/m^2$) By Downwind Sectors
Segment Boundaries In Miles

Direction From Site	0.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	8.694E-09	2.686E-09	1.069E-09	5.841E-10	3.712E-10	1.594E-10	4.944E-11	1.960E-11	1.046E-11	6.477E-12
SSW	5.762E-09	1.780E-09	7.084E-10	3.871E-10	2.460E-10	1.057E-10	3.277E-11	1.299E-11	6.936E-12	4.293E-12
SW	4.749E-09	1.467E-09	5.839E-10	3.191E-10	2.028E-10	8.710E-11	2.701E-11	1.071E-11	5.717E-12	3.538E-12
WSW	4.125E-09	1.274E-09	5.071E-10	2.771E-10	1.761E-10	7.565E-11	2.346E-11	9.298E-12	4.965E-12	3.073E-12
W	4.855E-09	1.500E-09	5.969E-10	3.262E-10	2.073E-10	8.905E-11	2.761E-11	1.094E-11	5.844E-12	3.617E-12
WNW	4.502E-09	1.391E-09	5.534E-10	3.024E-10	1.922E-10	8.256E-11	2.560E-11	1.015E-11	5.419E-12	3.354E-12
NW	4.045E-09	1.250E-09	4.973E-10	2.718E-10	1.727E-10	7.420E-11	2.301E-11	9.119E-12	4.870E-12	3.014E-12
NNW	2.937E-09	9.072E-10	3.610E-10	1.973E-10	1.254E-10	5.386E-11	1.670E-11	6.619E-12	3.535E-12	2.188E-12
N	7.773E-09	2.402E-09	9.557E-10	5.222E-10	3.319E-10	1.426E-10	4.421E-11	1.752E-11	9.357E-12	5.792E-12
NNE	1.129E-08	3.487E-09	1.388E-09	7.583E-10	4.820E-10	2.070E-10	6.420E-11	2.544E-11	1.359E-11	8.410E-12
NE	9.103E-09	2.812E-09	1.119E-09	6.115E-10	3.887E-10	1.669E-10	5.177E-11	2.052E-11	1.096E-11	6.782E-12
ENE	4.908E-09	1.516E-09	6.033E-10	3.297E-10	2.095E-10	9.001E-11	2.791E-11	1.106E-11	5.907E-12	3.656E-12
E	6.899E-09	2.132E-09	8.482E-10	4.635E-10	2.946E-10	1.265E-10	3.924E-11	1.555E-11	8.305E-12	5.140E-12
ESE	9.195E-09	2.841E-09	1.130E-09	6.177E-10	3.926E-10	1.686E-10	5.230E-11	2.073E-11	1.107E-11	6.851E-12
SE	8.252E-09	2.550E-09	1.015E-09	5.544E-10	3.524E-10	1.514E-10	4.693E-11	1.860E-11	9.934E-12	6.149E-12
SSE	7.369E-09	2.277E-09	9.059E-10	4.950E-10	3.146E-10	1.351E-10	4.191E-11	1.661E-11	8.870E-12	5.490E-12

Figure 2.3-201 [Deleted]

2.4 Hydrology

2.4.1 Hydrologic Description

NAPS COL 2.0-12-A The information needed to address DCD COL Item 2.0-12-A is included in SSAR Section 2.4.1, which is incorporated by reference with the following supplements.

2.4.1.1 Site and Facilities

The second paragraph of this SSAR section is supplemented as follows with information on the site grade elevation for Unit 3 and the effects on site drainage.

NAPS COL 2.0-12-A The design plant grade elevation for Unit 3 safety-related structures is 88.4 m (290.0 ft) msl. Figure 2.1-201 shows the layout of the external structures and components of Unit 3. The layout of Unit 3 will affect a few small wetlands and the upstream portions of two intermittent streams that flow north into an unnamed arm of Lake Anna just northwest of the power-block area. These areas will be partially filled in for the construction of the Unit 3 cooling towers in the CIRC. The drainage in these areas will be redirected to drainage swales and storm water management basins before rejoining the two intermittent streams. There are no other natural drainage features requiring changes to accommodate Unit 3. Evaluations of the flood levels from various flooding sources as they relate to protection of safety-related facilities for Unit 3 are discussed in Sections 2.4.2 and 2.4.10.

2.4.2 Floods

NAPS COL 2.0-13-A The information needed to address DCD COL Item 2.0-13-A is included in SSAR Section 2.4.2, which is incorporated by reference with the following supplements.

2.4.2.2 Flood Design Considerations

The last paragraph of this SSAR section is supplemented as follows with information on the design plant grade elevation for Unit 3.

NAPS COL 2.0-13-A The design plant grade for Unit 3 safety-related components and structures is at Elevation 88.4 m (290.0 ft) msl providing 6.89 m (22.61 ft) of freeboard above the design basis flooding level.

2.4.2.3 Effects of Local Intense Precipitation

This SSAR section is supplemented as follows to show that local intense precipitation is discharged to Lake Anna and that safety-related structures are located at elevations above the maximum water surface elevation produced by local intense precipitation.

NAPS COL 2.0-13-A

The site layout, drainage facilities, and drainage areas are shown on Figure 2.4-201. The safety-related buildings, which consist of the reactor, control, and fuel buildings, are located in the center and along the high point of the power block. From the high point, the site grading falls at a 1 percent slope to drainage ditches located along the northern and southern edges of the power block. The north and south drainage ditches convey the collected runoff from the power block and surrounding areas as shown on Figure 2.4-201 to the plant storm water management basin located in the northeast corner of the site. The storm water management basin discharges to Lake Anna through a bio-retention under-drain and a riser and pipe outlet. An emergency spillway over the plant access road is also provided to discharge large storm events, such as the PMP peak discharge, to Lake Anna. In performing the runoff analysis for the PMP storm, the under-drain and riser pipe outlet were conservatively assumed to be clogged. The sub-basin drainage areas shown on Figure 2.4-201 are summarized in Table 2.4-201 and Table 2.4-202.

NAPS ESP COL 2.4-4

For typical design storm events, such as the 10-year storm, runoff from the plant area is conveyed to the north and south drainage ditches through catch basins and storm drains as shown on Figure 2.4-201. Both the north and south drainage ditches also pass through culverts at road crossings and through the switchyard area. For the PMP runoff analysis, however, all underground storm drains and culverts were conservatively assumed to be completely clogged. Therefore, all flows were assumed to be overland or in open ditches.

NAPS COL 2.0-13-A

The PMP runoff analysis was performed on the north and south drainage ditches to determine the peak water levels during the PMP event and compare them to the design plant grade elevations for the safety-related buildings. There are additional ditches in the northeast corner that convey runoff from the power block to the north ditch. However, during the PMP event, these ditches would be inundated by overflows from the north drainage ditch and they were not included in the PMP analysis.

The rational method was used to determine the peak discharges for each of the sub-basin drainage areas shown on Figure 2.4-201. Two runoff coefficients were selected to represent ground cover conditions in the sub-basins. Conservative coefficients were selected to represent saturated ground conditions and also to reflect the intense rainfall that would occur during a PMP event. For vegetated areas, a runoff coefficient of 0.9 was used. For all other areas, a runoff coefficient of 1.0 was used to reflect an impervious surface. Composite runoff coefficients were determined based on the percentage of vegetated and impervious land cover for each sub-basin outlet point. Time of concentration values were estimated for each sub-basin using Natural Resources Conservation Service methodologies (Reference 2.4-201). To account for the non-linear response for large storms such as the PMP, the estimated time of concentration values were reduced by 25 percent as per guidance from the U.S. Army Corps of Engineers (Reference 2.4-202). PMP rainfall intensities were developed from the values listed in SSAR Table 2.4.3 and are shown in Figure 2.4-202. Using a duration equal to the reduced time of concentration for each sub-basin, the PMP rainfall intensity for each sub-basin was determined from Figure 2.4-202. The PMP peak discharge for each sub-basin was determined using the sub-basin point of interest drainage area, runoff coefficient, and PMP rainfall intensity. The estimated values for each sub-basin are shown in Table 2.4-203.

The steady-state backwater method in the computer program HEC-RAS (Reference 2.4-203) was used to estimate the peak PMP water levels in the north and south drainage ditches. HEC-RAS was first used to model the PMP flows over the storm water basin emergency spillway and determine the peak PMP water level in the basin, which then became the starting water level at the downstream most cross sections for the north and south drainage ditches. Cross-section data for the storm water basin spillway (outfall) and the north and south drainage ditches are shown on Figure 2.4-203 and Table 2.4-204.

Plant access roads cross the north and south drainage ditches at three locations. At each of these locations, the culverts under the roads were assumed to be blocked for the PMP runoff analysis. Inline weirs were used in HEC-RAS to model the road crossings and the flow over the top of the roads.

Manning's roughness coefficients (n values) for the channel and over bank areas were assigned based on guidance provided by Chow (Reference 2.4-204). Ditch linings consist of both grass vegetation and rip rap. Manning's n values of 0.030 for grass lined ditches and 0.035 for rip rap lined ditches were used. Land cover in the ditch over bank areas consist of grass vegetation, gravel and pavement. The paved areas are usually small areas located in large gravel areas. Therefore, Manning's n values to describe pavement were not used and values describing gravel cover were used for paved areas. This is a conservative approach as Manning's n values for gravel cover are higher than those for paved areas and produce higher water levels. For the grass over bank areas, a value of 0.030 was used and a value of 0.035 was used for the gravel over bank areas.

The peak discharges listed in Table 2.4-203 were entered into the HEC-RAS model conservatively at the upstream end of each sub-basin. The results of the HEC-RAS analysis are summarized in Table 2.4-204.

NAPS ESP COL 2.4-5

The design plant grade elevation for safety-related structures is Elevation 88.4 m (290.0 ft) msl as shown in Figure 2.1-201. As shown in Table 2.4-204, all cross sections in the power block area have maximum water surface elevations below Elevation 88.4 m (290.0 ft) msl. The maximum PMP water level in the power block area is Elevation 87.54 m (287.2 ft) msl, which is 0.85 m (2.8 ft) below the design plant grade elevation for safety-related structures.

NAPS COL 2.0-13-A

At the eastern edge of the Unit 3 site where the plant access road crosses the south drainage ditch, the grade elevation at the high point between the Unit 3 site and the Units 1 and 2 site is at Elevation 82.98 m (272.25 ft) msl. The maximum water level at the inline weir is Elevation 82.94 m (272.1 ft) msl, which is 0.05 m (0.15 ft) below the high point elevation and thus all Unit 3 PMP flows will be confined to the Unit 3 site and runoff generated from Unit 3 will not impact the Units 1 and 2 site.

Grading in the vicinity of the safety-related structures slopes away from the individual structures such that PMP ground and roof runoff will sheet flow away from each of these buildings and towards the collection ditches preventing flood flows from entering the buildings. Some ponding may occur near storm drain inlets and other depressed areas. The ponding will be temporary, however, and limited to the depressed areas. No storm drain inlets or depressed areas are located near safety-related buildings.

The Unit 3 site drainage facilities and grading in the power block area provide evacuation of the runoff from the PMP storm event. The design plant grade elevations for safety-related buildings are located above the estimated PMP water levels and grading is such that sheet flows and roof drainage flow away from safety-related buildings. Additionally, the Unit 3 PMP flows do not impact the Units 1 and 2 site. No flood protection measures are necessary for the Unit 3 site.

2.4.3 Probable Maximum Flood on Streams and Rivers

NAPS COL 2.0-14-A

The information needed to address DCD COL Item 2.0-14-A is included in SSAR Section 2.4.3 which is incorporated by reference with the following supplements.

The third paragraph of this SSAR section is supplemented as follows with information on the design plant grade elevation for Unit 3 safety-related facilities.

The design basis flooding elevation at the Unit 3 site is 81.50 m (267.39 ft) msl. This elevation is 6.89 m (22.61 ft) below the Unit 3 design plant grade elevation of 88.4 m (290.0 ft) msl for safety-related facilities, including the reactor building, which contains the safety-related UHS SSCs. Also, the Fire Water Service Complex (FWSC), which provides an on-site source of water supply to the UHS is at the same grade elevation as the reactor building. The FWSC components are above the design plant grade elevation and are therefore above the design basis flooding elevation. Because the site grade and access to the connection on Unit 3 for supply of make-up water to the UHS are above the design basis flooding elevation, the water supply to the UHS is capable of withstanding the PMF on streams and rivers without loss of the UHS safety functions.

2.4.4 Potential Dam Failures

NAPS COL 2.0.15-A

The information needed to address DCD COL Item 2.0-15-A is included in SSAR Section 2.4.4, which is incorporated by reference with the following supplements.

The second paragraph in this SSAR section is supplemented as follows to address the ESBWR UHS design.

NAPS ESP COL 2.4-6
NAPS ESP COL 2.4-7

DCD Section 9.2.5 describes the UHS and addresses NRC requirements to provide sufficient emergency cooling capability. The UHS for the passive ESBWR design is in the reactor building and does not use safety-related engineered underground reservoirs or storage basins. The service water system is not safety-related for the ESBWR. Even if Lake Anna were to be drained due to a dam failure, no safety-related structures or systems for Unit 3 would be adversely affected.

2.4.5 Probable Maximum Surge and Seiche Flooding

NAPS COL 2.0-16-A

The information needed to address DCD COL Item 2.0-16-A is included in SSAR Section 2.4.5, which is incorporated by reference.

2.4.6 Probable Maximum Tsunami Flooding

NAPS COL 2.0.17-A

The information needed to address DCD COL Item 2.0-17-A is included in SSAR Section 2.4.6, which is incorporated by reference.

2.4.7 Ice Effects

NAPS COL 2.0-18-A

The information needed to address DCD COL Item 2.0-18-A is included in SSAR Section 2.4.7, which is incorporated by reference with the following supplements.

2.4.7.2 Description of the Cooling Water System

The second paragraph of this SSAR section is supplemented as follows with information on the emergency cooling system for Unit 3.

NAPS COL 2.0-18-A

The emergency cooling water for Unit 3 is provided from the UHS as described in DCD Section 9.2.5.

The last paragraph of this SSAR section is supplemented as follows with information on normal and emergency cooling system functions for Unit 3 specific systems.

The normal cooling systems for Unit 3 are nonsafety-related systems. The emergency cooling system for Unit 3 is provided by the UHS, described in DCD Section 9.2.5, which is not affected by ice conditions. There is no safety-related system interconnection or inter-system reliance between normal and emergency cooling.

2.4.7.4 Frazil Ice

The fifth paragraph of this SSAR section is supplemented as follows with information on site-specific design for Unit 3.

NAPS COL 2.0-18-A The design of the Unit 3 intake is such that approach velocities are less than 0.5 fps. The SSAR stated that flow less than 1 fps would not produce sufficient turbulence to generate frazil ice. While this low flow may not produce sufficient turbulence to generate frazil ice, based on criteria stated in SSAR Reference 27 and others, there are other extreme climate factors that could combine and could cause formation of such ice. However, the Plant Service Water System (PSWS), which uses pumps in the Unit 3 intake for water make-up, is not safety-related. Information on the UHS is found in DCD Section 9.2.5.

The last paragraph of this SSAR section is supplemented as follows with information on preventing possible effects of anchor ice on the Unit 3 intake.

The most likely location for anchor ice to form is at the intake trash racks or intake screens. In the event of shutdown of Units 1 and 2 during cold weather, continuous rotation of traveling water screens and use of the trash removal rake on the intake trash rack will be effective in preventing any anchor ice formation.

2.4.7.5 Surface Ice

The second paragraph of this SSAR section is supplemented as follows with information on preventing possible effects of surface ice on the Unit 3 intake structure.

NAPS COL 2.0-18-A

Additionally, the skimmer wall at the front of the Unit 3 pump intake structure extends below the design low water level to further preclude the entry of ice sheets.

The fourth paragraph of this SSAR section is supplemented as follows with information showing emergency cooling for Unit 3 is not affected by surface ice formation.

Ice forces are accounted for in the design of the Unit 3 intake structure. It should also be noted that the intake and associated pumps for Unit 3 do not perform safety-related functions. The PSWS is supplied by pumps in the intake structure, but this system is not safety-related. Emergency cooling needed during a DBA is supplied by a separate UHS as discussed in DCD Section 9.2.5. Therefore, no safety-related Unit 3 facilities are affected by ice layer formation on the lake.

The last paragraph of this SSAR section is supplemented as follows with information showing emergency cooling for Unit 3 is not affected by the break-up of surface ice.

The presence of the skimmer wall, trash racks and traveling screens at the Unit 3 intake prevent ice floes from reaching the pumps. The accumulation of ice at the trash racks and traveling screens could clog them and reduce the flow capacity of the intake structure. However, since the PSWS is not safety-related and emergency cooling is provided by the UHS, no safety-related facilities are affected by ice floe accumulation on the lake.

2.4.7.6 Ice and Snow Roof Loads on Safety Related Structures

The last paragraph of this SSAR section is supplemented as follows with information to show ice and snow loads for Unit 3 safety-related structures are accounted for in the design.

NAPS COL 2.0-18-A Acceptable roofing structure performance for each safety-related roof is described in DCD Appendix 3G, e.g., for the Reactor Building, see DCD Section 3G.1.5.

2.4.8 Cooling Water Canals and Reservoirs

NAPS COL 2.0-19-A The information needed to address DCD COL Item 2.0-19-A is included in SSAR Section 2.4.8, which is incorporated by reference with the following supplements.

The third paragraph in this SSAR section is supplemented with information as follows to address whether Lake Anna is used for safety-related water withdrawals.

NAPS ESP COL 2.4-8 The UHS for Unit 3 is described in DCD Section 9.2.5. The IC/PCCS pools have their own water in place during Unit 3 operation for safety-related cooling in the event that use of the UHS is required. The North Anna Reservoir and Waste Heat Treatment Facility (WHTF), which comprise Lake Anna, are not used for safety-related water withdrawal for Unit 3.

2.4.9 Channel Diversions

NAPS COL 2.0-20-A The information needed to address DCD COL Item 2.0-20-A is included in SSAR Section 2.4.9, which is incorporated by reference.

2.4.10 Flooding Protection Requirements

NAPS COL 2.0-21-A

The information needed to address DCD COL Item 2.0-21-A is included in SSAR Section 2.4.10, which is incorporated by reference with the following supplements.

The first paragraph of this SSAR section is supplemented as follows with information on the site grade elevation for Unit 3.

The design plant grade is at Elevation 88.4 m (290.0 ft) msl (a greater height above the maximum design basis Lake Anna flood level of 81.5 m (267.39 ft) msl than was assumed in the SSAR).

The first paragraph of this SSAR section is further supplemented as follows with information to address slope embankment protection features for the Unit 3 intake structure.

NAPS ESP COL 2.4-9

The Unit 3 station water intake structure pump house is located in a separate intake channel west of the cove that houses the intake structure pump house for Units 1 and 2 as shown on Figure 2.4-204. The Unit 3 intake channel area is separated from Lake Anna by an outer berm constructed in the early 1980s. The top of the outer berm is Elevation 77.7 m (255 ft) msl and protects the Unit 3 intake channel area from flood events up to the 100-year flood on Lake Anna, which has an estimated flood level at Elevation 77.7 m (255.0 ft) msl (SSAR Reference 23). Flow from Lake Anna passes though a multi-barrel culvert in the outer berm as shown on Figure 2.4-204. The Unit 3 make-up water intake structure pump house and the intake channel area are protected from wind wave activity on Lake Anna by the outer berm, which has no visible indications of erosion or damage from wave activity. Rip-rap protection of the slope embankment at the pump house location is provided to prevent local runoff from eroding the embankment near this on-shore intake structure. It should be noted that although protection is provided, the Unit 3 make-up water intake structure pump house is not a safety-related structure.

The second paragraph of this SSAR section is supplemented as follows with information to show that flood protection measures are not required for the Unit 3 site.

NAPS COL 2.0-21-A

A local PMP drainage analysis was performed assuming, conservatively, that all underground storm drains and culverts are clogged. Details of the local PMP analysis and the resulting flood levels are presented in Section 2.4.2.3. The maximum PMP water level in the power block area is predicted to be at Elevation 87.5 m (287.2 ft) msl, which is 0.9 m (2.8 ft) below Elevation 88.4 m (290.0 ft) msl, the design plant grade elevation for safety-related facilities. Thus, no Unit 3 safety-related structure is subject to static or dynamic loading due to flooding as a result of design basis flood events or local PMP events. No flood protection measures are required for the Unit 3 site. Additionally, no technical specifications or emergency procedures are required to implement flood protection activities.

2.4.11 Low Water Considerations

NAPS COL 2.0-22-A

The information needed to address DCD COL Item 2.0-22-A is included in SSAR Section 2.4.11, which is incorporated by reference with the following supplements.

2.4.11.5 Plant Requirements

This SSAR section is supplemented as follows with information on the operational modes for the circulating water cooling system (CIRC) with respect to low water conditions.

NAPS ESP COL 2.4-10

The Unit 3 CIRC operates in either of two operating modes:

- Energy Conservation (EC)—The dry cooling array is bypassed and cooling water is circulated directly to the hybrid tower with a provision for cold weather bypass.
- Maximum Water Conservation (MWC)—The dry cooling tower and hybrid cooling tower operate in series with a provision for cold weather bypass.

Generally, when the North Anna Reservoir water level is at or above Elevation 76.2 m (250 ft) msl at the dam, and adequate reservoir discharge is being maintained, the EC mode is used. However, if the reservoir water level falls below Elevation 76.2 m (250 ft) msl and is not restored within a reasonable period of time, the MWC mode is used. While in the MWC mode, the dry tower fans may be turned off to provide additional electrical output during hours of peak demand.

As discussed in Section 2.4.14, Unit 3 will be shut down when the water level in Lake Anna drops below Elevation 73.762 m (242.0 ft) msl.

2.4.11.6 Heat Sink Dependability Requirements

This SSAR section is supplemented as follows with information on the effect of low water conditions on the UHS.

NAPS COL 2.0-22-A The Unit 3 UHS is described in DCD Section 9.2.5. Lake Anna is not relied on as a safety-related source of water withdrawals for emergency cooling.

2.4.12 Groundwater

NAPS COL 2.0-23-A The information needed to address DCD COL Item 2.0-23-A is included in SSAR Section 2.4.12, which is incorporated by reference with the following supplements and variances.

2.4.12.1.2 Local Hydrogeology

The third paragraph of this SSAR section is supplemented as follows based on additional borings.

NAPS COL 2.0-23-A Borings drilled as part of the ESP subsurface investigation program (SSAR Appendix 2.5.4B) and the Unit 3 subsurface investigation program (Appendix 2.5.4AA) penetrated saprolite to depths ranging from about 1.52 m (5 ft) to 24.99 m (82 ft). The saprolite penetrated by these borings is classified as a micaceous, silty-clayey, fine to coarse sand or sandy silt, with occasional (less than 10 percent) to some (between 10 and 50 percent) rock fragments.

The fifth paragraph of this SSAR section is supplemented as follows with information on additional groundwater level measurements data.

Groundwater at the Unit 3 site occurs in unconfined conditions in both the saprolite and underlying bedrock. The results of previous investigations at the site indicate that a hydrologic connection exists between the saprolite and the bedrock. (SSAR Reference 45) This condition has been confirmed as part of the ESP and Unit 3 subsurface investigation programs (SSAR Appendix 2.5.4B and Appendix 2.5.4AA) by the presence of nearly equal water level elevations recorded in the following observation well pairs: OW-845 and OW-846; OW-841 and OW-951; OW-848 and OW-950; and OW-842 and OW-949. (Figure 2.4-205). The wells are installed adjacent to each other, one sealed in the bedrock and the other in the saprolite. Water level elevations are provided in Table 2.4-15R. At the Unit 3 site, the water table is considered to be a subdued reflection of the ground surface and, therefore, the direction of groundwater movement is toward areas of lower elevations (SSAR Reference 45). Measurements made between December 2002 and May 2007 in observation wells at the site exhibit water level elevations ranging from about Elevation 72.54 m (238 ft) msl (relative to NAVD88) to Elevation 95.70 m (314 ft) msl, with corresponding ground surface elevations of about Elevation 86.25 m (283 ft) and Elevation 102.11 m (335 ft) msl, respectively (Table 2.4-15R). The measurements shown in Table 2.4-15R characterize short-term seasonal variability in the site water levels. Figure 2.4-205 presents hydrographs based on the water levels provided in this table for the 16 observation wells (OW-841 through OW-849, OW-901, OW-945 through OW-947, and OW-949 through OW-951) installed during the ESP and Unit 3 subsurface investigation programs and three monitoring wells (P-10, P-14, and P-18) previously installed for Units 1 and 2. The other wells being monitored (P- and WP-) were installed previously for Units 1 and 2 groundwater monitoring purposes around the SWR and the ISFSI, respectively. Figure 2.4-206 shows the locations of the observation wells.

Piezometric head contour maps (Figure 2.4-207 through Figure 2.4-214), prepared using water levels measured from December 2002 through May 2007 (Table 2.4-15R), indicate that groundwater flow is generally to the north and east, toward Lake Anna. Freshwater Creek and Elk Creek, both of which flow to Lake Anna, form hydrologic boundaries to the west and south of the site, respectively (SSAR Reference 46). Because the water levels in the observation wells are generally above the top of the well screen, the water level elevation represents the piezometric head. An evaluation of the piezometric head contours shown on Figure 2.4-207 through Figure 2.4-214, and using the maximum groundwater level observed in OW-901 (Elevation 88.08 m (289 ft) msl) and the minimum level observed in OW 848 (Elevation 73.76 m (242 ft) msl), with a distance between the two wells of 346.86 m (1,138 ft), results in a calculated hydraulic gradient toward Lake Anna of about 1.22 m (4 ft) per 30.48 m (100 ft).

NAPS ESP VAR 2.0-3

The eighth paragraph of this SSAR section is supplemented as follows with information on hydraulic conductivity values.

Thirteen groundwater observation wells installed at the site as part of the ESP and Unit 3 subsurface investigation programs were tested using the slug test method to determine hydraulic conductivity values for the saprolite and underlying shallow bedrock (SSAR Appendix 2.5.4B and Appendix 2.5.4AA). In addition, borehole packer tests were conducted in the bedrock at one of the Unit 3 observation well locations (OW-949) as an alternate method for determining hydraulic conductivity in the bedrock. Hydraulic conductivities calculated for the saprolite, based on tests in eleven wells, range from 0.076 to 3.017 m/day (0.25 to 9.9 ft/day), with a geometric mean of 0.53 m/day (1.74 ft/day). The hydraulic conductivity of the shallow bedrock, as determined from tests in two wells, is estimated to range from 0.152 to 1.920 m/day (0.5 to 6.3 ft/day) with a geometric mean of 0.625 m/day (2.05 ft/day). Table 2.4-16R summarizes the hydraulic conductivity data.

NAPS ESP VAR 2.0-2

The ninth paragraph in this SSAR section is supplemented as follows with information on additional geotechnical data and calculations of void ratio and total porosity.

Bulk densities for the bedrock range from 23.56 kN/m^3 (150 pounds per cubic foot) (pcf)) for highly to moderately weathered rock to 25.76 kN/m^3 (164 pcf) for slightly weathered to fresh rock (Table 2.5-212). Laboratory tests to determine the moisture content of saprolite samples indicate a median moisture content of about 17 percent (Table 2.5-212). Laboratory tests to determine the specific gravity of saprolite samples indicate a median specific gravity of 2.65 (Appendix 2.5.4AA). Using the median moisture content of 17 percent and a value of 2.65 for the specific gravity of the saprolite, the void ratio of the saprolite is estimated to be about 0.45. The void ratio is defined as the ratio of the volume of the voids to the volume of the solids and for a fully saturated soil is calculated as follows (Reference 2.4-205):

NAPS ESP VAR 2.4-1

$$\text{Void Ratio} = \text{moisture content} \times \text{specific gravity}$$

Using a void ratio of 0.45 for the saprolite, the total porosity is estimated to be about 31 percent. The porosity is defined as the ratio of the volume of the voids to the total volume of the soil. The void ratio and porosity are inter-related as follows (Reference 2.4-205):

Total Porosity = void ratio / (1 + void ratio)

Using a total porosity of 0.31, an effective porosity of about 25 percent is estimated based on 80 percent of the total porosity.

The tenth paragraph of this SSAR section is supplemented as follows with information on calculations of seepage velocity and travel time.

Based on the estimated hydraulic gradient, hydraulic conductivity, and effective porosity indicated above, groundwater beneath the Unit 3 site is expected to flow toward Lake Anna at a rate of about 0.085 m/day (0.28 ft/day). This groundwater seepage velocity is calculated as follows (Reference 2.4-206):

Seepage Velocity = (hydraulic conductivity × hydraulic gradient) / effective porosity

Travel time is defined as the time it takes the groundwater to move a set distance and is calculated as follows:

Travel Time = distance / velocity

Using a distance of approximately 304.8 m (1000 ft) between the Unit 3 radwaste building and the closest point along the shoreline of Lake Anna, the groundwater travel time is estimated to be about 10 years.

2.4.12.1.3 Plant Groundwater Use

The first paragraph of this SSAR section is supplemented as follows with information on the number and allocation of water supply wells at the site.

NAPS COL 2.0-23-A

Groundwater withdrawal for use by Units 1 and 2 is accomplished from three water supply wells permitted for public use by the Virginia Department of Health (VDH). These three wells (Nos. 4 (new), 6, and 7) comprise a single water supply system at the site. A separately permitted North Anna Nuclear Information Center (NANIC) well provides the water supply for the NANIC, while a fifth well provides water to the security training building. A sixth well is used to supply water to the Metrology/Environmental laboratory. Two other site wells (Number 2 and old Number 4) are not normally used, but are available, if needed. Well Number 3A is scheduled to be closed in accordance with Virginia regulations. The locations of these wells are shown on Figure 2.4-215 and the wells are described in Table 2.4-17R.

The second paragraph of this SSAR section is supplemented as follows with information on the individual and total capacities of the primary groundwater supply system wells for Units 1 and 2 (Nos. 4 (new), 6, and 7).

The three wells comprising the primary groundwater supply system for Units 1 and 2 have individual capacities ranging from 0.166 to 0.235 m^3/min (44 to 62 gpm) and a total capacity of 0.609 m^3/min (161 gpm). These three wells are permitted by the VDH for a total design capacity of 487.56 m^3/min (128,800 gpd), or about 0.337 m^3/min (89 gpm), based on a determination of the wells' capacity to supply an equivalent population of 3680 employees. Well Number 2 has a reported capacity of 0.034 m^3/min (9 gpm) and old Number 4 has a reported capacity of 0.204 m^3/min (54 gpm). (Reference 2.4-207)

The third paragraph of this SSAR section is supplemented as follows with information on the monthly groundwater withdrawal quantities of the primary groundwater supply system wells for Units 1 and 2 (Nos. 4 (new), 6, and 7).

Table 2.4-205 shows the monthly withdrawal quantities that were reported for the year ending December 31, 2006. It can be determined from this table that the primary wells withdrew a combined average of almost 0.027 m^3/min (7.25 gpm) for the year, and that the NANIC well withdrew an average of a little over 0.0038 m^3/min (1 gpm). The highest total monthly withdrawal in 2006 for the combined wells averaged almost 0.053 m^3/min (14 gpm) in March. (Reference 2.4-208)

The fourth paragraph of this SSAR section is supplemented as follows with information to explicitly state that groundwater is not used for safety-related purposes.

Any groundwater supply required by Unit 3 will not be used for safety-related purposes and will come from the existing wells or from drilling additional wells.

2.4.12.3 Monitoring of Safeguard Requirements

The fifth and sixth paragraphs of this SSAR section are supplemented as follows with information on the groundwater monitoring program required during and following construction of the plant.

NAPS COL 2.0-23-A

Because the Units 1 and 2 groundwater monitoring wells were not considered to be of sufficient areal extent to determine groundwater levels beneath the Unit 3 site, nine additional observation wells were installed as part of the ESP subsurface investigation program and seven additional observation wells were installed as part of the Unit 3 subsurface investigation program. Water levels in these 16 wells and 10 of the Units 1 and 2 monitoring wells (Table 2.4-15R) were measured between December 2002 and May 2007 to provide data on groundwater flow direction, gradient, and seasonal groundwater level fluctuations at the site.

Prior to site earthwork activities, some observation wells will need to be closed. As discussed in Section 2.5.4.5.1, the design plant grade elevation for Unit 3 is 88.4 m (290 ft). To achieve this elevation, excavation will be required in the southern portion of the power block area while lower areas to the north will need to be filled. As a result, existing observation wells in these and other areas of the site will be closed prior to the start of earthwork activities. An evaluation of the existing observation well locations will be performed to determine which wells will be closed and if any new wells will be required to establish an adequate monitoring network for the evaluation of impacts on site groundwater levels during plant construction. Closed wells will be grouted in compliance with Virginia regulations.

Evaluation of the groundwater monitoring program will include a review of the frequency with which groundwater level measurements are made in the observation wells. Groundwater levels in all or selected wells will be measured on a monthly basis for the duration of any temporary dewatering activities, and on a quarterly basis thereafter for two years following the completion of construction. Groundwater levels will then be measured on a semi-annual or annual basis during plant operation.

2.4.12.4 Design Bases for Subsurface Hydrostatic Loading

The first paragraph of this SSAR section is supplemented as follows with information on the design plant grade elevation for Unit 3.

NAPS COL 2.0-23-A

This maximum groundwater level means that a permanent dewatering system is not needed for safe operation of Unit 3, based on the groundwater design bases for safety-related SSCs as described in DCD Section 3.4.1 and the comparison with the DCD site parameter value for maximum groundwater level as shown in Table 2.0-201.

The third paragraph of this SSAR section is supplemented as follows with information on the maximum groundwater level for hydrostatic loading purposes.

Construction of Unit 3 at a design plant grade elevation of 88.4 m (290 ft), 5.8 m (19 ft) higher than that of Units 1 and 2, will result in the maximum groundwater level in this area being higher than that previously estimated in the SSAR. The pre-construction ground surface in the Unit 3 power block area ranges in elevation from about 96.93 m (318 ft) (B-919) to 82.91 m (272 ft) (B-928) and the piezometric head contour maps (Figure 2.4-207 through Figure 2.4-214) indicate that groundwater level elevations in this area range from about 91.44 to 80.77 m (300 to 265 ft).

As discussed in Section 2.5.4.5.1, the Unit 3 design plant grade elevation will be achieved by excavation in the southern portion of the power block area and filling in lower areas to the north. A 3-horizontal to 1-vertical (3H:1V) slope will be cut into the existing natural ground surrounding the southern and eastern sides of the plant area.

Because earthwork and construction associated with Unit 3 will alter the existing groundwater levels within the power block area, a numerical groundwater flow model was constructed to evaluate these effects and determine maximum post-construction groundwater levels beneath the power block area. The groundwater model was developed using site-specific hydrogeologic and hydrologic data and the computer code Visual MODFLOW Pro 4.2 (Reference 2.4-209). The post-construction piezometric head contour map (Figure 2.4-216) indicates that maximum groundwater level elevations in the power block area range from about 82.60 to 86.26 m (271 to 283 ft). Therefore, the maximum groundwater level elevation in the power block area of Unit 3 is 86.26 m (283 ft) or

2.134 m (7 ft) below the design plant grade elevation of 88.4 m (290 ft). This maximum groundwater level means that a permanent dewatering system is not needed for safe operation of Unit 3, based on the groundwater design bases for safety-related SSCs as described in DCD Section 3.4.1 and the comparison with the DCD site parameter value for maximum groundwater level as shown in Table 2.0-201.

2.4.13 Accidental Releases of Liquid Effluents to Ground and Surface Waters

NAPS COL 2.0-24-A

The information needed to address DCD COL Item 2.0-24-A is included in SSAR Section 2.4.13, which is incorporated by reference with the following supplements.

Mitigating design features considered acceptable by BTP 11-6 (Reference 2.4-210) are incorporated into the design of Unit 3 to preclude an accidental release of liquid effluents. Descriptions of these features are provided below.

Below-grade tanks containing radioactivity are located on levels B1F and B2F of the Radwaste Building. The Radwaste Building is designed to seismic requirements as specified in DCD Table 3.2-1. In addition, compartments containing high level liquid radwaste are steel lined up to a height capable of containing the release of all liquid radwaste in the compartment. Releases as a result of major cracks in tanks result in the release of the liquid radwaste to the compartment and then to the building sump system for containment in other tanks or emergency tanks. Because of these design capabilities, it is considered remote that any major event involving the release of liquid radwaste into these volumes results in the release of these liquids to the groundwater environment via the liquid pathway.

The Condensate Storage Tank (CST), part of the Condensate Storage and Transfer System (CS&TS), is the only above-grade tank that contains radioactivity outside of containment. The CS&TS, described in DCD Section 9.2.6, meets GDC 60 by compliance with RG 1.143, Position C.1.2 for design features provided to control the release of liquid effluents containing radioactive material. The basin surrounding the tank is designed to prevent uncontrolled runoff in the event of a tank failure. The basin volume is sized to contain the total tank capacity. Tank overflow is also collected in this basin. A sump located inside the retention basin has provisions for sampling collected liquids prior to routing them to the Liquid Waste Management System (LWMS) or the storm sewer as per sampling and release requirements. These design features are intended to preclude the release of liquids from the CST to either the ground or surface water environment via the liquid pathway.

NAPS ESP PC 3.E(3)

The mitigating design features described above demonstrate that the radioactive waste management systems, structures, and components for Unit 3, as defined in RG 1.143, include features to preclude accidental releases of radionuclides into potential liquid pathways. Nevertheless, in accordance with SRP 11.2, an analysis of accidental releases of radioactive liquid effluents in groundwater and surface water is performed. Descriptions and results of these analyses are provided herein.

The source term provided in DCD Table 12.2-13a, Liquid Waste Management System Equipment Drain Collection Tank Activity, is used in the analysis of an accidental release of liquid effluents from an equipment drain collection tank and the radwaste building structure to the groundwater system. This source term is appropriate because these tanks collect radioactive liquids from various pieces of plant equipment and are upstream of liquid processing by the LWMS.

The CST is used as the source in the analysis of an accidental release of liquid effluent to the surface water system. The radionuclide concentrations expected to be present in the CST are as given in Table 2.4-212.

2.4.13.1 Groundwater

The purpose of this section is to provide a conservative analysis of a postulated, accidental release of radioactive liquid effluents to the groundwater at the Unit 3 site. The accident scenario is described. The model used to evaluate radionuclide transport is presented, along with potential pathways of contamination to water users. The radionuclide transport analysis is described, and the results are summarized. The radionuclide concentrations to which a water user might be exposed are compared against the regulatory limits.

2.4.13.1.1 Accident Scenario

A liquid radwaste tank outside of containment is postulated to rupture with its contents released to the groundwater. The volume of the liquid assumed to be released and the associated radionuclide concentrations were selected to produce an accident scenario that leads to the most adverse contamination of groundwater, or surface water via the groundwater pathway.

Radwaste tanks outside of containment are located on the levels B1F and B2F of the radwaste building as shown on DCD Figure 1.2-25. The radwaste tanks having the largest volumes include the three equipment drain collection tanks and the equipment drain sample tank, all in the lowest level, B2F. Each of these tanks has a volume of 140 m^3 (37,000 gal) according to DCD Tables 12.2-13a and 12.2-13b.

Estimates of activity concentrations in various liquid radwaste tanks are provided in DCD Tables 12.2-13a through 12.2-13g. Of these tanks, the limiting tank in terms of radionuclide activity is the Equipment Drain Collection Tank, and its activity is provided in DCD Table 12.2-13a. Values are also provided in Table 2.4-206.

The accident scenario assumes that one of the equipment drain collection tanks ruptures and its contents are released to the groundwater. Note that this accident scenario is extremely conservative because the radwaste building is seismically designed in accordance with RG 1.143, Class RW-IIa, as described in DCD Section 12.2.1.4. Also, the concrete in each tank cubicle is provided with a steel liner, as described in Section 11.2.2.3, to prevent any potential liquid releases to the environment.

2.4.13.1.2 **Model**

Figure 2.4-217 illustrates the model used to evaluate an accidental release of radioactive liquid effluent to groundwater, or to surface water via the groundwater pathway. The key elements and assumptions embodied in the model are described and discussed below.

As indicated above, one of the equipment drain collection tanks is assumed to be the source of the release, with each tank having a capacity of 140 m^3 (37,000 gal) and radionuclide concentrations as given in DCD Table 12.2-13a. These tanks are located on the lowest level of the radwaste building (level B2F), which has a floor elevation of 244 ft msl. One of the tanks is postulated to rupture, and 80 percent of the liquid volume (112 m^3 or 29,600 gal) is assumed to be released following the guidance provided in BTP 11-6. Following tank rupture, it is conservatively assumed that a pathway is created that allows the entire 112 m^3 to enter the groundwater (unconfined aquifer) instantaneously.

The assumption of instantaneous release to the groundwater following tank rupture is very conservative because it requires failure of the floor drain system, plus it ignores the barriers presented by the basemat and the steel liners incorporated into the tank cubicles of the radwaste building, which is seismically designed. It should also be recognized that level B2F of the radwaste building is well below the water table. Piezometric head contour maps presented in Figure 2.4-207 through Figure 2.4-214 indicate that the ambient water table in the vicinity of the radwaste building is about 270 ft msl, or 26 ft above the floor elevation. If the basemat or exterior walls of the radwaste building and associated steel liners were to fail simultaneously, groundwater would flow into the radwaste building, precluding the release of liquid effluents out of the building. Only if the interior of the radwaste building was flooded to a level higher than the surrounding groundwater would there be a pathway for liquid effluents to be released out of the building and to the groundwater. Hence, the assumption of an accidental release of liquid effluents from the radwaste building to groundwater is extremely conservative, given the design features of the radwaste building intended to prevent an accidental release and the hydrogeologic conditions at the site.

With the postulated instantaneous release of the contents of an equipment drain collection tank to groundwater, radionuclides enter the unconfined aquifer and migrate with the groundwater in the direction of decreasing hydraulic head. Hydraulic head contour maps for the unconfined aquifer presented in Figure 2.4-207 through Figure 2.4-214 indicate that the groundwater pathway from the radwaste building is north-northeast toward Lake Anna, a groundwater discharge area. In particular, the hydrogeologic data suggest that the groundwater pathway terminates in the cove used for the Unit 3 intake from Lake Anna. The flow path is assumed to be a straight line between the radwaste building and the south edge of the cove, a distance of about 305 m (1000 ft) based on Figure 2.1-201. As indicated in Section 2.4.12.1.2, groundwater flow occurs in both the saprolite and underlying, shallow bedrock. During saturated zone transport, radionuclide concentrations of the liquid released to the groundwater are reduced by the processes of adsorption, hydrodynamic dispersion, and radioactive decay. As described in Section 2.4.12.1.3, there is an existing water-supply well in the power block area (Well No. 2 on Figure 2.4-215). This well will be closed and grouted to accommodate the construction of Unit 3. There are no other existing water-supply or monitoring wells between the postulated release point and Lake Anna.

Lake Anna serves as a groundwater discharge area for the unconfined aquifer. The radionuclides associated with a liquid release enter the surface water system via Lake Anna. As noted above, the portion of Lake Anna closest to the release point is the cove that is used for the water supply intake for Unit 3. This cove was created to construct the intakes for two units (an earlier Unit 3 and a Unit 4) that were not completed. The water-supply intake for Unit 3 is located at the end of the cove, which serves as the forebay for Unit 3's water-supply intake. This cove is separated from the rest of the lake by a cofferdam. Openings in the cofferdam are provided to convey water from the North Anna Reservoir to the water-supply intake. This intake provides make-up water to the normal plant circulating water and service water cooling systems, and supplies water to the potable, demineralized, and fire protection water systems. The make-up water flow rates are about 1.42 m^3/s (50 cfs) and about 0.96 m^3/s (34 cfs) in the energy conservation and maximum water conservation modes, respectively. Under normal operating conditions, any contaminated groundwater discharging to the cove is entrained, mixed, and diluted with surface water in the Unit 3 intake forebay area and subsequently abstracted from the cove by the water-supply intake for Unit 3. Any radionuclides introduced into the make-up water systems are either circulated through the closed-cycle, wet cooling towers associated with the normal plant circulating water and service water cooling systems, or enter the potable, demineralized, and fire protection systems. Volatile radionuclides in the circulating water passing through wet cooling towers are lost to the atmosphere. Non-volatile radionuclides concentrate in the circulating water due to evaporative losses and are discharged with the cooling tower blowdown to the discharge canal. The blowdown discharge, about 0.34 m^3/s (12 cfs) in energy conservation mode and about 0.25 m^3/s in maximum water conservation mode, mixes in the discharge canal with 120 m^3/s (4246 cfs) of circulating water from Units 1 and 2 as illustrated in SSAR Figure 2.4-13. Radionuclides transported by the flow in the discharge canal then pass through the WHTF, enter the North Anna Reservoir through Dike 3, and undergo additional mixing and dilution in the reservoir (SSAR Figure 2.4-13). Most of the flow and associated dissolved radionuclide constituents are then recirculated upstream to the water intakes for Units 1, 2 and 3, while a relatively small fraction of the flow discharges from the North Anna Dam to the North Anna River.

As described in SSAR Section 2.1.1.3, the liquid effluent release limits for Unit 3 apply at the end of the discharge canal, which is designated the release point to unrestricted areas in the context of 10 CFR 20. As noted in ESP-ER Table 2.3-4, the Doswell Water Treatment Plant is the nearest and only municipal water system currently supplied from the North Anna River. The treatment plant is about 20 miles downstream of the North Anna Dam and near the confluence with the Little River.

2.4.13.1.3 Radionuclide Transport Analysis

A radionuclide transport analysis has been conducted to estimate the radionuclide concentrations that might expose existing and future water users based on an instantaneous release of the radioactive liquid from an equipment drain collection tank. Analysis of liquid effluent release commences with a screening model, using demonstratively conservative assumptions and coefficients. Radionuclide concentrations resulting from the screening analysis are then compared against the effluent concentration limits (ECLs) identified in 10 CFR 20, Appendix B, Table 2, Column 2, to determine acceptability. Further analysis, using more realistic modeling techniques, is conducted, as necessary, after the screening results for each step are available.

This analysis accounts for the parent radionuclides assumed present in the radwaste tank plus progeny radionuclides that are generated subsequently during transport. The analysis considered all progeny in the decay chain sequences that are important for dosimetric purposes. International Commission on Radiation Protection (ICRP) Publication 38 (Reference 2.4-211) was used to identify the member for which the decay chain sequence can be truncated. For some of the radionuclides assumed present in an equipment drain collection, consideration of up to three members of the decay chain sequence was required. The derivation of the equations governing the transport of the parent and progeny radionuclides follows.

Transport of the parent radionuclide along a groundwater pathline is governed by the advection-dispersion-reaction equation (Reference 2.4-212), which is given as:

$$R\frac{\partial C}{\partial t} = D\frac{\partial^2 C}{\partial x^2} - v\frac{\partial C}{\partial x} - \lambda RC \qquad (2.4.13\text{-}2)$$

where: C = radionuclide concentration; R = retardation factor; D = coefficient of longitudinal hydrodynamic dispersion; v = average linear velocity; and λ = radioactive decay constant. The retardation factor is defined from the relationship:

$$R = 1 + \frac{\rho_b K_d}{n_e} \qquad (2.4.13\text{-}3)$$

where: ρ_b = bulk density; K_d = distribution coefficient; and n_e = effective porosity. The average linear velocity is determined using Darcy's law, which is:

$$v = -\frac{K}{n_e}\frac{dh}{dx} \qquad (2.4.13\text{-}4)$$

where: K = hydraulic conductivity; and dh/dx = hydraulic gradient. The radioactive decay constant can be written as:

$$\lambda = \frac{\ln 2}{t_{1/2}} \qquad (2.4.13\text{-}5)$$

where: $t_{1/2}$ = radionuclide half-life.

Using the method of characteristics approach described in Reference 2.4-213, the material derivative of concentration can be written as:

$$\frac{dC}{dt} = \frac{\partial C}{\partial t} + \frac{dx}{dt}\frac{\partial C}{\partial x} \qquad (2.4.13\text{-}6)$$

Conservatively neglecting hydrodynamic dispersion, the characteristic equations for Equation 2.4.13-2 can be expressed as follows:

$$\frac{dC}{dt} = -\lambda C \qquad (2.4.13\text{-}7)$$

$$\frac{dx}{dt} = \frac{v}{R} \qquad (2.4.13\text{-}8)$$

The solutions of the system of equations comprising Equation 2.4.13-7 and Equation 2.4.13-8 can be obtained by integration to yield the characteristic curves of Equation 2.4.13-2. For the parent radionuclide, the equations representing the characteristic curves can be obtained as:

$$C_1 = C_{10}\exp(-\lambda_1 t) \qquad (2.4.13\text{-}9)$$

$$t = R_1 L / v \qquad (2.4.13\text{-}10)$$

where: C_1 = concentration of the parent radionuclide; C_{10} = initial concentration of the parent radionuclide; λ_1 = radioactive decay constant for the parent radionuclide; R_1 = retardation factor for the parent radionuclide; and L = groundwater pathline length.

Similar relationships exist for progeny radionuclides. For the first progeny in the decay chain, the advection-dispersion-reaction equation is:

$$R_2 \frac{\partial C_2}{\partial t} = D \frac{\partial^2 C_2}{\partial x^2} - v \frac{\partial C_2}{\partial x} + d_{12} \lambda_1 R_1 C_1 - \lambda_2 R_2 C_2 \tag{2.4.13-11}$$

where: subscript 2 denotes the first progeny radionuclide; and d_{12} = fraction of parent radionuclide transitions that result in production of progeny radionuclide. The characteristic equations for Equation 2.4.13-11, again conservatively neglecting hydrodynamic dispersion, can be derived as:

$$\frac{dC_2}{dt} = d_{12} \lambda_1' C_1 - \lambda_2 C_2 \tag{2.4.13-12}$$

$$\frac{dx}{dt} = \frac{v}{R_2} \tag{2.4.13-13}$$

where: $\lambda_1' = \lambda_1 R_1 / R_2$. Recognizing that Equation 2.4.13-12 is formally similar to Equation B.43 of Reference 2.4-214, these equations can be integrated to yield:

$$C_2 = K_1 \exp(-\lambda_1' t) + K_2 \exp(-\lambda_2 t) \tag{2.4.13-14}$$

$$t = R_2 L / v \tag{2.4.13-15}$$

for which:

$$K_1 = \frac{d_{12} \lambda_2 C_{10}}{\lambda_2 - \lambda_1'}$$

$$K_2 = C_{20} - \frac{d_{12} \lambda_2 C_{10}}{\lambda_2 - \lambda_1'}$$

The advection-dispersion-reaction equation for the second progeny in the decay chain is:

$$R_3 \frac{\partial C_3}{\partial t} = D \frac{\partial^2 C_3}{\partial x^2} - v \frac{\partial C_3}{\partial x} + d_{13} \lambda_1 R_1 C_1 + d_{23} \lambda_2 R_2 C_2 - \lambda_3 R_3 C_3 \tag{2.4.13-16}$$

where: subscript 3 denotes the second progeny radionuclide; d_{13} = fraction of parent radionuclide transitions that result in production of second progeny radionuclide; and d_{23} = fraction of first progeny radionuclide transitions that result in production of second progeny radionuclide. The characteristic equations for Equation 2.4.13-16, again conservatively neglecting hydrodynamic dispersion, can be derived as

$$\frac{dC_3}{dt} = d_{13}\lambda_1' C_1 + d_{23}\lambda_2' C_2 - \lambda_3 C_3 \qquad (2.4.13\text{-}17)$$

$$\frac{dx}{dt} = \frac{v}{R_3} \qquad (2.4.13\text{-}18)$$

where: $\lambda_1' = \lambda_1 R_1 / R_3$; and $\lambda_2' = \lambda_2 R_2 / R_3$. Considering the formal similarity of Equation 2.4.13-17 to Equation B.54 of Reference 2.4-214, Equation 2.4.13-17 and Equation 2.4.13-18 can be integrated to yield:

$$C_3 = K_1 \exp(-\lambda_1' t) + K_2 \exp(-\lambda_2' t) + K_3 \exp(-\lambda_3 t) \qquad (2.4.13\text{-}19)$$

$$t = R_3 L / v \qquad (2.4.13\text{-}20)$$

for which:

$$K_1 = \frac{d_{13}\lambda_3 C_{10}}{\lambda_3 - \lambda_1'} + \frac{d_{23}\lambda_2' d_{12}\lambda_3 C_{10}}{(\lambda_3 - \lambda_1')(\lambda_2' - \lambda_1')}$$

$$K_2 = \frac{d_{23}\lambda_3 C_{20}}{\lambda_3 - \lambda_2'} - \frac{d_{23}\lambda_2' d_{12}\lambda_3 C_{10}}{(\lambda_3 - \lambda_2')(\lambda_2' - \lambda_1')}$$

$$K_3 = C_{30} - \frac{d_{13}\lambda_3 C_{10}}{\lambda_3 - \lambda_1'} - \frac{d_{23}\lambda_3 C_{20}}{\lambda_3 - \lambda_2'} + \frac{d_{23}\lambda_2' d_{12}\lambda_3 C_{10}}{(\lambda_3 - \lambda_1')(\lambda_3 - \lambda_2')}$$

To estimate the radionuclide concentrations in groundwater discharging to Lake Anna, Equation 2.4.13-9, Equation 2.4.13-14, and Equation 2.4.13-19 were applied as appropriate along the groundwater pathline that would originate at the radwaste building and terminate at the Lake Anna shoreline. The analysis was performed sequentially as described below.

a. **Transport Considering Radioactive Decay Only**

An initial screening analysis was performed considering radioactive decay only. This analysis assumed that all radionuclides migrate at the same rate as groundwater and considered no adsorption and retardation, which would otherwise result in lower radionuclide concentrations. The concentrations of the radionuclides assumed to be released from an equipment drain collection tank were decayed for a period equal to the groundwater travel time from the point of release to Lake Anna, using

Equation 2.4.13-9, Equation 2.4.13-14, or Equation 2.4.13-19 as appropriate with $R_1 = R_2 = R_3 = 1$. Any radionuclide having a concentration of less than 0.01 times its ECL was eliminated from consideration because its concentration would be well below its regulatory limit. Any radionuclide having a concentration greater than or equal to 0.01 times its ECL was retained for further evaluation.

Evaluating transport considering radioactive decay only requires an estimate of the groundwater travel time. The groundwater travel time between the radwaste building and Lake Anna has been estimated using the following site-specific hydrogeologic characteristics:

K = 3.4 ft/day

dh/dx = 0.040 ft/ft

$n_e = 0.25$

Note that the hydraulic conductivity (3.4 ft/day) was established as a site characteristic in the SSAR based on hydraulic testing of 13 observation wells completed in the water table aquifer, with the 3.4 ft/day value being the maximum of the 13 observations. Subsequently, three additional observation wells were installed and tested as part of the Unit 3 subsurface investigation, increasing the total number of hydraulic conductivity observations to 16 for the saprolite material. Table 2.4-16R summarizes these data. A review of these observations indicates that 14 out of the 16 values (87.5 percent) are less than or equal to 3.4 ft/day. Because a value of 3.4 ft/day is greater than 87.5 percent of the observed data, it is considered to be a conservative value. The two values that exceed 3.4 ft/day include those observed at OW-945 (3.8 ft/day) and OW-946 (9.9 ft/day), which are located 2000 to 2500 ft upgradient from the radwaste building (see Figure 2.4-206). These values are not representative of the hydrogeologic conditions along the groundwater pathway between the radwaste building and Lake Anna. Hence, the 3.4 ft/day value established as site characteristic is conservative and is considered appropriate for assessing an accidental release of liquid effluent to the groundwater.

Using the above values in Equation 2.4.13-4, the average linear velocity is calculated to be:

$$v = -\frac{K}{n_e}\frac{dh}{dx} = \frac{3.4}{0.25} \times 0.040 = 0.544 \text{ ft/day}$$

The groundwater travel time is then:

$$t = L / v = 1000 / 0.54 = 1840 \text{ days} = 5.03 \text{ years}$$

Using Equation 2.4.13-9, Equation 2.4.13-14, or Equation 2.4.13-19 as appropriate with R = 1, the initial concentrations were decayed for a period of 5.03 years. Radioactive decay data and decay chain specifications were taken from NUREG/CR-5512, Vol. 1, Table E.1 (Reference 2.4-214). Radioactive decay data for some of the shorter-lived radionuclides were obtained from Reference 2.4-211. Table 2.4-206 summarizes the results and identifies those radionuclides for which the ratio of groundwater concentration to ECL would exceed 0.01. These include H-3, Mn-54, Fe-55, Co-60, Ni-63, Zn-65, Sr-90, Y-90, Ru-106, Ag-110m, Cs-134, Cs-137, Ce-144, and Pu-239.

b. **Transport Considering Radioactive Decay and Adsorption**

Radionuclides retained from the screening analysis (H-3, Mn-54, Fe-55, Co-60, Ni-63, Zn-65, Sr-90, Y-90, Ru-106, Ag-110m, Cs-134, Cs-137, Ce-144, and Pu-239) were further evaluated and screened considering adsorption and retardation in addition to radioactive decay. Distribution coefficients for these elements were assigned using literature values. In particular, K_d values were selected assuming the literature data to be log-normally distributed and selecting the 10th percentile of the distribution to conservatively assign a value for the radionuclide transport analysis. NUREG/CR-6697 (Reference 2.4-215), Attachment C, Table 3.9-1 is used to assign the mean and standard deviation for each of the distributions. In the case of Y-90, no data were available to assign a K_d value for yttrium. Instead, adsorption characteristics for yttrium were assumed to be similar to that of scandium, as these two elements lie adjacent in the periodic table. The K_d value for Y-90 was then estimated as the 10th percentile of the distribution for scandium using the mean and standard deviation from NUREG/CR-6697.

To assess the validity of the Kd values derived from NUREG/CR-6697 as described above, site-specific Kd values were determined for Mn, Fe, Co, Ni, Zn, Sr, Ru, Ag, Cs, Ce, and Pu for 20 saprolite and weathered rock samples. These samples were obtained from borings B-901, B-904, B-913, B-917, B-919, B-920, B-928, B-929, B-931, B-932, B-949, and B-951, the locations of which are shown on Figures 2.5-221 and 2.5-222. Kd values for these samples were determined using the batch method in accordance with ASTM D 4646-03 at Savannah River National

Laboratory using site water obtained from the unconfined aquifer. Table 2.4-207 summarizes the results along with the values estimated from NUREG/CR-6697. Comparing the site-specific Kd values against those assumed in the transport analysis indicates the following:

- The Kd values assumed for 6 elements (Fe, Zn, Sr, Ru, Cs, Ce) are less than the minimum observed values.
- The Kd values assumed for 2 elements (Mn, Co) are bounded by the 1 percentile of the observed data.
- The Kd values assumed for 2 elements (Ag, Pu) are bounded by the 10th percentile of the observed data.
- The Kd value assumed for 1 element (Ni) is bounded by the 25th percentile of the observed data.

Based on the above comparison, the Kd values derived from NUREG/CR-6697 are conservative relative to the site-specific values. The literature values were therefore retained for the transport analysis.

Retardation factors were calculated using Equation 2.4.13-3 with the distribution coefficients established as described above, an effective porosity of 0.25, and a bulk density of 1.83 g/cm^3. The bulk density was estimated using a soil grain specific gravity of 2.65 and total porosity of 0.31, as described in Section 2.4.12.1.2. The concentration for each radionuclide was then determined at the point of groundwater discharge to Lake Anna using Equation 2.4.13-9 or Equation 2.4.13-14 and the appropriate initial concentration, decay rate, and retardation factor. Results are summarized in Table 2.4-208 and indicate that groundwater concentration to ECL ratios for H-3, Sr-90, Y-90, and Pu-239 would exceed 0.01.

c. **Transport Considering Radioactive Decay, Adsorption, and Dilution**

As discussed in Section 2.4.13.1.2, the H-3, Sr-90, Y-90, and Pu-239 isotopes discharging with the groundwater to Lake Anna are entrained, mixed, and diluted in the surface water, flow of which is induced by the water-supply intake for Unit 3. A dilution factor was estimated to account for the mixing and dilution as described below.

The total radionuclide flux in the groundwater was calculated using NUREG-0868 (Reference 2.4-216), Equation 3.23 as a basis.

Conservatively ignoring hydrodynamic dispersion, this equation can be restated as:

$$F_{GW} = n_e v C_{GW} A \quad (2.4.13\text{-}21)$$

where: F_{GW} = total radionuclide flux in groundwater; C_{GW} = radionuclide concentration in the groundwater; A = cross-sectional area normal to the direction of groundwater flow; and the other terms are as defined previously. The cross-sectional area of the plume is conservatively assumed to extend over the entire saturated thickness of the unconfined aquifer and the entire length of the radwaste building. The saturated thickness is taken to extend from the water table to the top of the Zone III-IV, slightly weathered to moderately weathered rock. In the vicinity of the radwaste building, Figure 2.4-207 through Figure 2.4-214 indicate a water table elevation of about 82.30 m (270 ft) msl, while Table 2.5-208 indicates the Zone III-IV top of rock elevation to be 74.37 m (244 ft) msl. These values result in a saturated thickness of about 7.92 m (26 ft). DCD Figure 1.2-25 indicates the radwaste building to be 65 m (213 ft) in length normal to the direction of groundwater flow. The assumption that the plume extends the entire length of the building is conservative because the characteristic dimensions of the sources from which a release is postulated are a relatively small fraction of the 65 m length. The cross-sectional area is then the product of 26 ft and 213 ft, or 5540 ft^2.

The total radionuclide flux in the surface water of Lake Anna, induced by pumping from the water-supply intake for Unit 3, is calculated as:

$$F_{SW} = Q C_{SW} \quad (2.4.13\text{-}22)$$

where: F_{SW} = total radionuclide flux in surface water; Q = surface water flow rate; and C_{SW} = radionuclide concentration in the surface water. This approach for calculating the radionuclide flux in surface water is justified, considering that any radionuclides released to the groundwater would likely discharge to the Unit 3 intake forebay area, which has been isolated from the rest of the lake and from which the water intake for Unit 3 will obtain water. The surface water flow is determined by the water supply requirements for Unit 3, which total 1.42 m^3/s (50 cfs) when running in the energy conservation mode and 0.96 m^3/s (34 cfs) in the maximum water conservation mode. There are times of the year when the combination wet and dry cooling towers used for normal plant cooling could function in a completely dry mode, particularly during cold weather. Under these conditions, no make-up water is required for the normal plant circulating water system, which comprises most of the total demand. However, these conditions are expected to persist for relatively short durations and are not representative of transport conditions over longer time scales.

Because the total radionuclide flux must be conserved, radionuclide concentrations in the surface water are estimated by equating Equation 2.4.13-21 and Equation 2.4.13-22 and solving for C_{SW}:

$$C_{SW} = \frac{n_e v A}{Q} C_{GW} \qquad (2.4.13\text{-}23)$$

where the quantity $n_e vA/Q$ defines the dilution factor. Assuming for conservatism that the plant is operating in the maximum water conservation mode, the dilution factor is calculated using the previously defined values for n_e, v, A, and Q to be:

$$\frac{n_e v A}{Q} = \frac{0.25 \times 0.54 / 86{,}400 \times 5540}{34} = 2.56 \times 10^{-4}$$

This dilution factor is applied to the H-3, Sr-90, Y-90, and Pu-239 concentrations reported in Table 2.4-209 to account for dilution in addition to radioactive decay and adsorption. Table 2.4-210 summarizes the resulting concentrations, which represent the concentrations in the surface water withdrawn by the water-supply intake for Unit 3. It is seen that the concentrations of each of these radionuclides are below their respective ECLs.

Most of the 0.96 m^3/s (34 cfs) withdrawn from Lake Anna is used as make-up water to replenish evaporative losses from cooling towers that are part of closed-cycle cooling systems. As discussed in Section 2.4.13.1.2, the non-volatile radionuclides concentrate in the circulating water by a factor of about four, prior to being discharged to the discharge canal. Even then, concentrations are well below ECLs. It should also be noted that radionuclides released in cooling tower blowdown discharge would mix with circulating water discharge from Units 1 and 2 (up to 120.2 m^3/s (4246 cfs)) as long as these units are operating. If Units 1 and 2 are shutdown, a minimum of 15.04 m^3/s (531 cfs) will continue to be circulated to provide adequate dilution for normal plant releases. These flows from Units 1 and 2 would further dilute the radionuclides discharged from Unit 3, which is not accounted for in Table 2.4-209.

As described in Section 2.4.13.1.2, there is an atmospheric pathway associated with the accidental release of liquid effluents to groundwater, which entails the release of tritium to the atmosphere, as water vapor, from the evaporation of cooling water from the Unit 3 wet cooling towers. Table 2.4-209 indicates a tritium concentration of 5.08×10^{-7} μCi/cm^3 (508 pCi/l) for surface water withdrawn as makeup water to the circulating water system and contributed by the accidental release. This value is about one-twentieth the 9417 pCi/l value evaluated previously in FEIS Appendix H.3 (Reference 2.4-217). The FEIS determined that the doses associated with a concentration of 9417 pCi/l were insignificant when compared to the maximally exposed individual (MEI) dose from atmospheric releases from the stacks of Unit 3. Because the predicted concentration of 508 pCi/l is about a factor of twenty less than 9417 pCi/l, the dose associated with this atmospheric pathway is also insignificant.

2.4.13.1.4 **Compliance with 10 CFR 20**

The radionuclide transport analysis presented above demonstrates that each of the radionuclides that could be accidentally released to groundwater would be individually below its ECL. However, 10 CFR 20, Appendix B, Table 2, imposes additional requirements when the identity and concentration of each radionuclide in a mixture are known. In this case, the ratio present in the mixture and the concentration otherwise established in 10 CFR 20, Appendix B for the specific radionuclide not in a mixture must be determined. The sum of such ratios for all of the radionuclides in the mixture may not exceed “1” (i.e., “unity”).

This sum of fractions approach was applied to the radionuclide concentrations conservatively estimated as described in Section 2.4.13.1.3. Results are summarized in Table 2.4-210. The ratios for the mixture sum to 5.64×10^{-2}. This value is multiplied by a factor of four to account for concentration of radionuclides in circulating water due to evaporative losses, which results in a value of 2.26×10^{-1}. This value is below unity and demonstrates that an accidental release of radioactive liquid effluent in groundwater complies with the 10 CFR 20 limits at the entrance of the discharge canal. The 2.26×10^{-1} value is bounding because the 0.25 m^3/s (9 cfs) of blowdown discharge would be diluted with a minimum of 15.04 m^3/s (531 cfs) of flow from Units 1 and 2 within the discharge canal and prior to the end of the canal, which is designated as the release point to unrestricted areas.

2.4.13.2 **Surface Water**

The purpose of this section is to provide a conservative analysis of a postulated, accidental release of radioactive liquid effluents to the surface water at the Unit 3 site. The key assumptions and accident scenario are described. The dilution analysis is presented along with various plant operating scenarios. The bounding case is identified. The radionuclide concentrations to which a water user might be exposed are compared against the regulatory limits for the bounding case.

2.4.13.2.1 **Assumptions**

The key assumptions adopted in this analysis area are as follows:

- The accidental release of radioactive liquid effluents to surface water results from a failure of the CST.
- The radionuclide inventory for the CST is based on 80 percent of the volume capacity of that tank as recommended in BTP 11-6. Based on the CST capacity of 4885 m^3 (172,512 ft^3) given in DCD Table 9.2-10, the volume of liquid released is 3908 m^3 (138,010 ft^3).
- The containment dike surrounding the CST fails simultaneously, allowing the liquid contents of the CST to enter the Stormwater Retention Pond 1, which discharges to the North Anna Reservoir as shown in Figure 2.1-201.
- The discharge canal behaves as a fully mixed system.
- The liquid effluent release limits established in 10 CFR 20 apply at the end of the discharge canal, which is designated as the release point to unrestricted areas in accordance with SSAR Section 2.1.1.3.

2.4.13.2.2 **Accident Scenario**

Figure 2.1-201 illustrates the locations of the plant facilities and hydrologic features involved in an accidental liquid release of liquid effluent to surface water from a failure of the CST.

With the postulated release of the contents of the CST and concurrent failure of the CST containment dike, the liquid effluent would enter the storm drain system and collect in Stormwater Retention Pond 1. The outlet from Pond 1 discharges to the North Anna Reservoir just outside the forebay area for the Unit 3 intake. This forebay area is separated from the North Anna Reservoir by a cofferdam (the outer berm) that was constructed for the abandoned Units 3 and 4. The intake channel culvert installed through the cofferdam conveys water from the reservoir to the forebay.

Depending on the operating status of Units 1, 2, and 3, liquid effluent discharged from Pond 1 to the North Anna Reservoir can be entrained in the circulating water intakes for Units 1 and 2, or entrained in the Unit 3 intake. When Units 1, 2, and 3 are operating normally or Unit 3 is operating by itself, the discharge from Pond 1 is assumed to be entrained in the make-up water flow for Unit 3 due to the proximity of the Pond 1 outfall to the culvert entrance to the Unit 3 intake forebay. When the circulating water pumps for Units 1 and 2 are operating and Unit 3 is shutdown, the Pond 1 discharge is assumed to be entrained in the circulating water flow for Units 1 and 2.

For the cases in which liquid effluent is entrained in the make-up water flow for Unit 3, radionuclides introduced into the make-up water system are circulated through the closed-cycle, wet cooling towers associated with the normal plant circulating water and service water cooling systems. Volatile radionuclides in the circulating water passing through wet cooling towers are lost to the atmosphere, while any radionuclides remaining in solution are subject to drift loss to the atmosphere and subsequent deposition. Non-volatile radionuclides concentrate in the circulating water due to the evaporative losses and are discharged with the cooling tower blowdown to the discharge canal. This blowdown discharge mixes in the discharge canal with circulating water flow discharge from Units 1 and 2. In the event that Unit 3 is not operating and liquid effluent is entrained by the circulating water flow for Units 1 and 2, radionuclides enter the once-through circulating water system and enter the discharge canal with the circulating water flow.

2.4.13.2.3 **Dilution Analysis**

Based on the accident scenario described above, the liquid effluent resulting from a failure of the CST and its containment dike would be entrained by the intake structures for Units 1 and 2 or Unit 3, circulated through their respective wet cooling systems, and released to the discharge canal. Depending on plant operating statuses, four alternative dilution scenarios are possible, which are described below. It is conservatively assumed that Unit 3 operates in the maximum water conservation mode, as opposed to the energy conservation mode, because of the lower dilution potential of the maximum water conservation mode.

1. **Units 1, 2, and 3 normal operation** - All three units are operating at capacity. The combined circulating water flow rate for Units 1 and 2 is Q_{12} = 120.2 m^3/s (4246 ft^3/s) (Reference 2.4-218). The make-up flow rate for Unit 3, Q_{3MU}, is about 0.96 m^3/s (34 ft^3/s). The blowdown discharge rate for Unit 3, Q_{3BD}, is about 0.25 m^3/s (9 ft^3/s).

2. **Units 1 and 2 shutdown; Unit 3 normal operation** - Units 1 and 2 are shutdown, and Unit 3 is operating at capacity. For Units 1 and 2, a minimum of Q_{12} = 15.0 m^3/s (531 ft^3/s) is circulated to provide dilution of normal plant releases. The make-up flow rate for Unit 3, Q_{3MU}, is about 0.96 m^3/s (34 ft^3/s), and the blowdown discharge rate for Unit 3, Q_{BD}, is about 0.25 m^3/s (9 ft^3/s).

3. **Units 1 and 2 normal operation; Unit 3 shut down** - Units 1 and 2 are operating at capacity, and Unit 3 is shut down. The combined circulating water flow rate for Units 1 and 2 is Q_{12} = 120.2 m^3/s (4246 ft^3/s). The make-up and blowdown flow rates are zero for Unit 3.

4. **Units 1, 2, and 3 all shut down** - Units 1, 2, and 3 are all shut down. For Units 1 and 2, a minimum of Q_{12} = 15.0 m^3/s (531 ft^3/s) is circulated to provide dilution of normal plant releases. The make-up and blowdown flow rates are zero for Unit 3.

For scenarios 1 and 2 involving entrainment into the Unit 3 cooling system with subsequent release to the discharge canal, conservation of mass requires:

$$C_{3MU} = \frac{Q_{P1}}{Q_{3MU}} C_{CST} \tag{2.4.13-24}$$

$$C_{3BD} = NC_{3MU} \tag{2.4.13-25}$$

$$C_{DC} = \frac{Q_{3BD}}{Q_{3BD} + Q_{12}} C_{3BD} \tag{2.4.13-26}$$

where: C_{CST} = radionuclide concentration in CST; C_{3MU} = radionuclide concentration of make-up water entrained in Unit 3 intake;

C_{3BD} = radionuclide concentration in blowdown discharge water; C_{DC} = radionuclide concentration in discharge canal; N = number of cycles of concentration for the Unit 3 wet cooling towers; Q_{P1} = flow rate from Pond 1 into Lake Anna; Q_{3MU} = makeup water flow rate for Unit 3; and Q_{12} = circulating water flow rate for Units 1 and 2. For scenarios 3 and 4 involving entrainment into the circulating water system of Units 1 and 2 with subsequent release to the discharge canal, conservation of mass requires:

$$C_{DC} = \frac{Q_{P1}}{Q_{12}} C_{CST} \qquad (2.4.13\text{-}27)$$

Using the equations above, concentrations of a radionuclide released from the CST with a relative concentration of one (unity) are calculated for each of the alternative dilution scenarios described above. A value of N = 4 is assumed. A value of Q_{P1} = 0.017 m^3/s (0.60 ft^3/s) is used based on the outflow and storage characteristics of Stormwater Retention Pond 1. This value assumes a Pond 1 stage elevation of 79.90 m (262.13 ft) msl corresponding to 3908 m^3 (138,010 ft^3) of storage, which is the volume of liquid assumed to be released from the CST. Note that the radionuclide concentrations and discharge flow rate from Pond 1 assume that the pond is initially dry. If there were water in the pond prior to the CST failure, the radionuclide concentrations in the discharge would be more dilute and less conservative than those assumed. Table 2.4-211 summarizes the results. Of the various alternatives evaluated, scenario 2 produces the maximum relative concentration (1.18E-03) at the end of the discharge canal.

2.4.13.2.4 **Compliance with 10 CFR 20**

To determine regulatory compliance, the maximum relative concentration (1.18E-03) determined in Section 2.4.13.2.3 is used to scale the radionuclide concentrations assumed for the CST. Table 2.4-212 summarizes the results.

The results presented in Table 2.4-212 demonstrate that each of the radionuclides potentially released from the CST to surface water is below its ECL. However, 10 CFR 20, Appendix B, Table 2, imposes additional requirements when the identity and concentration of each radionuclide in a mixture are known. In this case, the ratio present in the mixture and the concentration otherwise established in 10 CFR 20, Appendix B for the specific radionuclide not in a mixture must be determined. The sum of such ratios for all of the radionuclides in the mixture may not exceed “1” (i.e., “unity”).

For the bounding scenario summarized in Table 2.4-212, the ratios sum to 1.7×10^{-1}. This value is below unity, demonstrating that an accidental liquid release of radioactive liquid effluent in surface water complies with 10 CFR 20 limits at the end of the discharge canal, which is designated as the release point to unrestricted areas.

NAPS COL 2.0-25-A

2.4.14 Technical Specifications and Emergency Operation Requirements

The design plant grade elevation for safety-related SSCs is located above the design basis flood level, as stated in Section 2.4.2, and above the maximum groundwater elevation, as stated in Section 2.4.12. Safety-related SSCs for the plant are protected from external floods as discussed in Section 3.4. The elevation of exterior access openings, which are above the PMF and local PMP flood levels, and the design of exterior penetrations below design flood and groundwater levels, which are appropriately sealed, result in a design and site combination that do not necessitate emergency procedures or meet the criteria for Technical Specification LCOs to ensure safety-related functions at the plant.

The plant elevation is also above flood and groundwater elevations for Regulatory Treatment of Non-Safety Systems (RTNSS) SSCs used to provide the makeup water to the UHS (IC/PCCS pools) from 72 hours to 7 days after an accident. The Seismic Category I FWSC SSCs are therefore also protected from external floods. Therefore, no technical specifications or emergency procedures are required to prevent hydrological phenomena from degrading the UHS.

NAPS ESP COL 2.4-2

Unit 3 will shutdown when the water level in Lake Anna drops below Elevation 73.762 m (242.0 ft) msl. Because this operational restriction is not related to protection of safety-related SSCs or degradation of the UHS, low lake level is not a Technical Specification LCO.

Section 2.4 References

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NAPS COL 2.0-23-A

Table 2.4-15R Quarterly Groundwater Level Elevations

Observation Well No.	Well Depth* (ft)	Reference Point Elev. (ft)	Reference Point Stickup** (ft)	Top of Well Screen Elev. (ft)	Well Screen Length (ft)	Groundwater Level Elevations Date of Measurement							
						12/17/02	03/17/03	06/17/03	09/29/03	02/01/05	11/29/06	02/28/07	05/30/07
OW-841	34.3	251.6	1.5	228.1	9.7	248.9	249.6	249.6	249.3	249.1	249.51	249.11	248.74
OW-842	49.6	336.7	1.5	297.8	9.6	307.5	308.9	310.8	312.0	314.2	313.36	313.84	314.23
OW-843	49.2	320.6	1.5	282.1	9.7	285.1	288.1	290.8	290.2	290.7	288.58	289.78	290.15
OW-844	24.6	273.5	1.5	257.6	9.6	265.5	266.7	267.3	266.4	266.2	266.49	266.32	265.63
OW-845	55.0	297.3	1.5	253.0	9.7	272.7	274.9	277.4	277.3	277.1	276.19	276.21	276.86
OW-846	32.7	297.3	1.5	273.5	9.8	272.5	274.8	277.1	277.0	276.8	276.01	275.95	276.59
OW-847	49.8	319.7	1.5	280.6	9.6	285.4	287.0	289.5	290.8	293.3	***	***	294.24
OW-848	47.3	284.5	1.5	240.8	5.0	241.7	242.9	243.6	244.0	243.2	243.86	243.2	242.63
OW-849	49.8	298.5	1.5	259.4	9.7	265.5	269.5	271.7	270.8	269.5	270.21	***	270.03
OW-901	108	311.3	1.70	214.6	10	N/A	N/A	N/A	N/A	N/A	285.13	286.98	288.46
OW-945	54.5	283.1	1.50	240.1	10	N/A	N/A	N/A	N/A	N/A	***	***	271.59
OW-946	43.4	335.6	1.60	303.6	10	N/A	N/A	N/A	N/A	N/A	302.86	302.8	312.62
OW-947	58.0	315.1	1.80	268.3	10	N/A	N/A	N/A	N/A	N/A	297.61	297.81	297.92
OW-949	104.5	336.9	1.23	243.2	10	N/A	N/A	N/A	N/A	N/A	313.69	313.9	314.39
OW-950	92.0	284.5	1.52	203.0	10	N/A	N/A	N/A	N/A	N/A	239.8	238.68	238.37
OW-951	67.1	250.7	1.01	194.6	10	N/A	N/A	N/A	N/A	N/A	249.44	249.6	249.4
P-10	22.5	286.4	2.4	267.0	5	274.4	274.8	275.2	275.2	275.3	275.48	275.4	275.17
P-14	N/A	327.1	N/A	N/A	N/A	271.6	272.2	272.8	273.1	273.8	273.99	274.03	274.09
P-18	N/A	329.0	N/A	N/A	N/A	285.7	286.5	287.5	288.4	289.9	290.48	290.72	290.9

NAPS COL 2.0-23-A

Table 2.4-15R Quarterly Groundwater Level Elevations

Observa-tion Well No.	Well Depth* (ft)	Reference Point Elev. (ft)	Reference Point Stickup** (ft)	Top of Well Screen Elev. (ft)	Well Screen Length (ft)	Groundwater Level Elevations Date of Measurement							
						12/17/02	03/17/03	06/17/03	09/29/03	02/01/05	11/29/06	02/28/07	05/30/07
P-19	58.5	322.3	N/A	N/A	5	284.3	285.2	286.3	287.3	288.9	***	***	290.46
P-20	61.0	320.6	N/A	N/A	5	274.9	275.4	275.8	275.0	276.7	277.1	276.95	276.95
P-21	58.5	319.2	N/A	N/A	5	Dry	261.2	262.0	262.4	263.4	263.74	263.65	263.88
P-22	60.0	320.5	N/A	N/A	5	276.8	277.8	278.6	278.9	279.5	279.79	279.58	279.45
P-23	41.2	296.4	1.9	258.7	5	261.1	262.6	263.3	263.1	263.5	263.56	263.34	263.35
P-24	25.0	293.4	2.3	271.3	5	276.4	277.1	278.4	278.3	278.4	278.82	278.8	278.08
WP-3	N/A	317.9(?)****	N/A	266.5	5	299.7	301.0	302.8	302.3	302.1	302.42	302.2	302.09
Lake Anna Water Level Elevation						248.1	250.1	250.4	250.1	250.1	250.1	250.1	249.8
Service Water Reservoir Water Level Elevation						314.6	313.3	314.6	314.6	314.5	314.5	314.4	314.5

OW-800 series wells installed in December 2002 as part of ESP Subsurface Investigation Program
OW-900 series wells installed in November 2006 as part of Unit 3 Subsurface Investigation Program
P- wells installed previously to monitor NAPS Units 1 and 2 Service Water Reservoir
WP- well installed previously as part of Interim Spent Fuel Storage Installation monitoring program
* Below ground surface at time of installation
** Above ground surface at time of installation
*** Valid reading not obtained.
**** Estimated elevation; not a survey result. See SSAR Appendix 2.5.4B.
N/A – not available

NAPS COL 2.0-23-A

Table 2.4-16R Hydraulic Conductivity Value

Observation Well No.	Depth Interval Tested (ft)	Elevation	Material	Hydraulic Conductivity cm/sec	Hydraulic Conductivity ft/day
PT-1 [a]	Near-surface	Unknown	Saprolite	2.8×10^{-5}	0.08
PT-2 [a]	Near-surface	Unknown	Saprolite	1.4×10^{-5}	0.04
P-10 [b]	14.5–22.5	269.5–261.5	Saprolite	6.1×10^{-4} to 6.1×10^{-5}	1.7 to 0.17
P-24 [b]	16.8–25.0	274.3–266.1	Saprolite	2.9×10^{-4} to 6.6×10^{-6}	0.8 to 0.02
P-23 [b]	33.7–41.2	260.7–253.2	Saprolite	6.6×10^{-5}	0.19
OW-844 [c]	12.7–24.6	259.3–247.4	Saprolite	9.9 to 8.9×10^{-5}	0.28 to 0.25
OW-841 [c]	20.1–34.3	230.0–215.8	Saprolite	8.2 to 7.8×10^{-4}	2.3 to 2.2
OW-846 [c]	20.3–32.7	275.5–263.1	Saprolite	1.2×10^{-3} to 6.8×10^{-4}	3.4 to 1.9
OW-847 [c]	35.0–49.8	283.2–268.4	Saprolite	2.3 to 2.1×10^{-4}	0.66 to 0.58
OW-842 [c]	35.3–49.6	299.9–285.6	Saprolite	3.3×10^{-4}	0.93
OW-849 [c]	35.6–49.8	261.4–247.2	Saprolite	1.1×10^{-3} to 7.0×10^{-4}	3.2 to 2.0
OW-843 [c]	36.4–49.2	282.7–269.9	Saprolite	4.9 to 4.5×10^{-4}	1.4 to 1.3
OW-848 [c]	39.1–47.3	243.9–235.7	Saprolite	1.2×10^{-3} to 9.9×10^{-4} [d]	3.4 to 2.8 [d]
OW-845 [c]	39.7–55.0	256.1–240.8	Quartz Gneiss	1.1×10^{-3} to 6.3×10^{-4} [e]	3.1 to 1.8 [e]
OW-945 [f]	41.5–51.5	240.1–230.1	Saprolite	1.4 to 1 x 10^{-3}	3.8 to 2.8
OW-946 [f]	30.4–40.4	303.6–293.6	Saprolite	3.5 to 2.6 x 10^{-3}	9.9 to 7.4
OW-947 [f]	45.0–55.0	268.3–258.3	Saprolite	2.4 to 1.6 x 10^{-4}	0.67 to 0.46
OW-949 [f]	92.5–102.5	243.2–233.2	Quartz Gneiss	8.4 to 6.7 x 10^{-4}	2.4 to 1.9
Packer Test Results					
B-949 [f]	84.0–89	250.8–245.8	Quartz Gneiss	1.7 x 10^{-4}	0.48
	94.5–99.5	240.3–235.3	Quartz Gneiss	2.2 x 10^{-3}	6.28

NAPS COL 2.0-23-A

Table 2.4-16R Hydraulic Conductivity Value

Observation Well No.	Depth Interval Tested (ft)	Elevation	Material	Hydraulic Conductivity cm/sec	Hydraulic Conductivity ft/day
Laboratory Test Results					
B-48 [a]	3.5	290.5	Sandy silt	1×10^{-6}	0.003
B-8 [a]	5.5	293.5	Fine sand, tr. silt	1×10^{-6}	0.003
B-2 [a]	15.5	269.5	Fine to med. sand, w/clayey silt	4×10^{-5}	0.11
B-15 [a]	36	281	Silty fine sand	1.3×10^{-5}	0.04

a. SSAR Reference 43
b. SSAR Reference 56
c. SSAR Appendix 2.5.4 B
d. Results may not be accurate due to static water level approximately 0.5 ft below top of well screen.
e. Results may not be accurate due to short duration of stable water level recovery measurements.
f. Appendix 2.5.4AA

NAPS COL 2.0-23-A
NAPS ESP VAR 2.4-2
ESP COR

Table 2.4-17R North Anna Power Station Water Supply Wells

Well	Depth (ft)	Measured Yield (gpd)	Design Yield (gpd)	Water Treatment
No. 2[a,b]	335	12,960	Unknown	Unknown (normally not in use)
No. 3A[a,b]	185	74,880		Unknown
No. 4 (new)[a,b]	305	63,360	35,200[c]	None
No. 6[a,b]	375	79,200	44,000[c]	None
No. 4 (old)[a,b] (not used)	200	77,760	NA	NA
NANIC[a,d]	260	106,560	19,600	Calcite filtration
Security Training Building	640	Unknown	Unknown	Unknown
No. 7[c]	730	89,280	49,600	None
Metrology Laboratory	116	Unknown	Unknown	Unknown

a. SSAR Reference 50
b. SSAR Reference 48
c. Reference 2.4-203
d. SSAR Reference 49

NAPS COL 2.0-13-A

Table 2.4-201 Unit 3 Sub-Basin Drainage Areas

Sub-Basin	Drainage Area (ft^2)	Drainage Area (acres)
B	334,935	7.67
S1	156,241	3.60
S2	100,005	2.30
S3	84,803	1.95
S4	384,081	8.82
N1	91,773	2.11
N2	181,035	4.16
N3	267,867	6.15
N4	168,076	3.86
N5	432,662	9.93
Total	2,201,478	50.55

NAPS COL 2.0-13-A

Table 2.4-202 Unit 3 Sub-Basin Point of Interest (POI) Drainage Areas

Sub-Basin	Contributing Upstream Sub-Basins	Total POI Drainage Area (acres)
B	All	50.55
S1	S1, S2, S3, S4	16.67
S2	S2, S3, S4	13.07
S3	S3, S4	10.77
S4	S4	8.82
N1	N1, N2, N3, N4, N5	26.21
N2	N2, N3, N4, N5	24.10
N3	N3, N4, N5	19.94
N4	N4	3.86
N5	N5	9.93

NAPS COL 2.0-13-A

Table 2.4-203 Unit 3 Site PMP Peak Discharges

Sub-Basin	POI Drainage Area (acres)	Composite Runoff Coefficient	Time of Concentration (min)	Rainfall Intensity (in/hr)	PMP Peak Discharge (cfs)
B	50.55	0.98	14.5	39.0	1932.0
S1	16.67	0.98	15.4	37.5	612.6
S2	13.07	0.97	14.6	39.0	494.4
S3	10.77	0.99	14.1	40.2	428.6
S4	8.82	0.99	13.0	42.5	371.1
N1	26.21	0.97	14.5	39.0	991.5
N2	24.10	0.97	13.8	40.8	953.8
N3	19.44	0.96	11.9	45.5	871.0
N4	3.86	0.97	10.7	50.0	187.2
N5	9.93	0.94	10.7	50.0	466.7

NAPS COL 2.0-13-A

Table 2.4-204 Unit 3 Site PMP Water Levels

Ditch	Cross Section	Discharge (cfs)	Maximum Water Level (ft)	Ditch/ Channel Bottom Width (ft)	Ditch/ Channel Invert El. (ft)	Bank El. (ft)
Outfall	630	1932.0	271.7	377	260.0	270.0
	565	1932.0	271.7	396	260.0	270.0
	425	1932.0	271.7	Weir	N/A	N/A
	300	1932.0	265.0	160	240.0	270.0
	0	1932.0	265.0	160	240.0	270.0
South	1774	371.1	287.0	4	282.0	286.0
	1720	371.1	286.9	4	281.8	286.0
	1570	371.1	286.6	4	281.6	286.0
	1512	371.1	286.4	4	281.5	286.0
	1414	371.1	286.3	4	281.4	286.0
	1365	371.1	286.1	4	281.3	286.0
	1317	371.1	286.0	4	281.2	286.0
	1265	371.1	285.8	4	281.2	286.0
	1177	371.1	285.5	4	281.0	284.0
	1063	428.6	284.9	4	280.8	284.0
	1013	428.6	284.5	4	280.6	284.0
	922	428.6	284.3	4	280.4	283.7
	820	494.4	282.7	4	280.0	281.4
	800	494.4	282.6	4	280.0	281.3
	782	485.7	282.1	4	280.0	281.2
	717	404.8	280.5	4	278.0	279.5
	615	338.4	278.4	4	276.3	277.5
	557	320.8	276.0	4	273.7	275.2
	497	320.8	273.9	4	271.7	273.1
	440	320.8	272.2	4	270.2	271.4

NAPS COL 2.0-13-A

Table 2.4-204 Unit 3 Site PMP Water Levels

Ditch	Cross Section	Discharge (cfs)	Maximum Water Level (ft)	Ditch/ Channel Bottom Width (ft)	Ditch/ Channel Invert El. (ft)	Bank El. (ft)
South	404	320.8	272.3	18.5	267.5	271.0
	380	320.8	272.1	Weir	N/A	N/A
	379	320.8	272.1	Weir	N/A	N/A
	332	320.8	272.1	8	266.2	271.0
	278	439.0	272.0	8	266.1	271.0
	195	612.6	271.8	8	266.0	271.0
North	1312	653.4	287.2	2	284.0	286.0
	1245	653.4	287.2	Weir	N/A	N/A
	1190	653.4	287.2	4	283.0	286.0
	1108	871.0	287.1	4	282.4	286.0
	987	871.0	287.1	4	281.5	284.0
	845	953.8	287.0	4	281.2	284.0
	802	953.8	286.8	4	281.2	284.0
	742	953.8	286.8	4	280.9	284.0
	662	953.8	286.7	4	280.8	284.0
	550	953.8	286.4	4	280.5	284.0
	500	953.8	286.4	Weir	N/A	N/A
	375	991.5	285.8	0	281.0	284.0
	288	991.5	284.7	0	280.1	283.2
	180	991.5	282.4	0	279.5	281.8
	90	991.5	277.7	0	273.7	278.1
	0	991.5	274.0	0	270.2	274.0
	-100	991.5	272.2	0	269.7	271.8

NAPS COL 2.0-23-A

Table 2.4-205 North Anna Power Station Groundwater Use [a] January 1, 2006 to December 31, 2006 (Millions of Gallons)

Month	Well #4 (new)	Well #6	Well #7
January	0.2545	0.0072	0
February	0.2895	0	0.0001
March	0.6233	0.0002	0.0002
April	0.0854	0.2029	0
May	0.0006	0.2901	0
June	0	0.3228	0
July	0.0013	0.3007	0.0001
August	0.0005	0.3933	0.0008
September	0.0763	0.2379	0
October	0.2123	0.0529	0
November	0.226	0.0311	0
December	0.1978	0.0081	0
Total	1.9675	1.8472	0.0012
Monthly Average	0.1640	0.1539	0.0001

a. Reference 2.4-208

NAPS COL 2.0-24-A

Table 2.4-206 Results of Groundwater Transport Analysis Considering Radioactive Decay Only

Parent Radionuclide	Progeny in Chain	Half-life[1] (days)	Branching Fraction[1]			Decay Rate[2] (days^{-1})	Collection Tank Conc[3] (MBq/m^{3})	Collection Tank Conc (μCi/cm^{3})	Ground Water Conc[4] (μCi/cm^{3})	ECL[5] (μCi/cm^{3})	Ground Water Conc / ECL
			d_{12}	d_{13}	d_{23}						
H-3		4.51E+03				1.54E-04	9.73E+01	2.63E-03	2.0E-03	1.00E-03	1.98E+00
Na-24		6.25E-01				1.11E+00	4.74E+01	1.28E-03	0.0E+00	5.00E-05	0.00E+00
P-32		1.43E+01				4.85E-02	1.98E+01	5.35E-04	1.1E-42	9.00E-06	1.20E-37
Cr-51		2.77E+01				2.50E-02	2.61E+03	7.05E-02	7.4E-22	5.00E-04	1.49E-18
Mn-54		3.13E+02				2.21E-03	9.83E+01	2.66E-03	4.5E-05	3.00E-05	1.51E+00
Mn-56		1.07E-01				6.48E+00	7.59E+01	2.05E-03	0.0E+00	7.00E-05	0.00E+00
Fe-55		9.86E+02				7.03E-04	3.08E+03	8.32E-02	2.3E-02	1.00E-04	2.29E+02
Fe-59		4.45E+01				1.56E-02	3.82E+01	1.03E-03	3.8E-16	1.00E-05	3.79E-11
Co-58		7.08E+01				9.79E-03	1.76E+02	4.76E-03	7.3E-11	2.00E-05	3.63E-06
Co-60		1.93E+03				3.59E-04	6.25E+02	1.69E-02	8.7E-03	3.00E-06	2.91E+03
Ni-63		3.51E+04				1.97E-05	3.24E+00	8.76E-05	8.4E-05	1.00E-04	8.44E-01
Cu-64		5.29E-01				1.31E+00	5.92E+01	1.60E-03	0.0E+00	2.00E-04	0.00E+00
Zn-65		2.44E+02				2.84E-03	2.65E+03	7.16E-02	3.9E-04	5.00E-06	7.73E+01
Rb-89		1.06E-02				6.54E+01	1.25E+00	3.38E-05	0.0E+00	9.00E-04	0.00E+00
	Sr-89	5.05E+01	1.0000			1.37E-02	1.43E+02	3.86E-03	4.3E-14	8.00E-06	5.33E-09
Sr-90		1.06E+04				6.54E-05	2.23E+01	6.03E-04	5.3E-04	5.00E-07	1.07E+03
	Y-90	2.67E+00	1.0000			2.60E-01	6.95E-01	1.88E-05	5.3E-04	7.00E-06	7.64E+01
Sr-91		3.96E-01				1.75E+00	5.68E+01	1.54E-03	0.0E+00	2.00E-05	0.00E+00
	Y-91m	3.45E-02	0.5780			2.01E+01			0.0E+00	2.00E-03	0.00E+00

NAPS COL 2.0-24-A

Table 2.4-206 Results of Groundwater Transport Analysis Considering Radioactive Decay Only

Parent Radionuclide	Progeny in Chain	Half-life[1] (days)	Branching Fraction[1]			Decay Rate[2] (days^{-1})	Collection Tank Conc[3] (MBq/m^3)	Collection Tank Conc (μCi/cm^3)	Ground Water Conc[4] (μCi/cm^3)	ECL[5] (μCi/cm^3)	Ground Water Conc / ECL
			d_{12}	d_{13}	d_{23}						
	Y-91	5.85E+01		0.4220	1.0000	1.18E-02	6.28E+01	1.70E-03	5.9E-13	8.00E-06	7.42E-08
Sr-92		1.13E-01				6.14E+00	3.25E+01	8.78E-04	0.0E+00	4.00E-05	0.00E+00
	Y-92	1.48E-01	1.0000			4.68E+00	2.67E+01	7.22E-04	0.0E+00	4.00E-05	0.00E+00
Y-93		4.21E-01				1.65E+00	5.98E+01	1.62E-03	0.0E+00	2.00E-05	0.00E+00
Zr-95		6.40E+01				1.08E-02	1.34E+01	3.62E-04	8.2E-13	2.00E-05	4.09E-08
	Nb-95m	3.61E+00	0.0070			1.92E-01			6.1E-15	3.00E-05	2.02E-10
	Nb-95	3.52E+01		0.9930	1.0000	1.97E-02	8.76E+00	2.37E-04	1.8E-12	3.00E-05	6.06E-08
Mo-99		2.75E+00				2.52E-01	2.07E+02	5.59E-03	3.3E-204	2.00E-05	1.67E-199
	Tc-99m	2.51E-01	0.8760			2.76E+00	1.72E+01	4.65E-04	3.2E-204	1.00E-03	3.23E-201
Ru-103		3.93E+01				1.76E-02	2.39E+01	6.46E-04	5.4E-18	3.00E-05	1.79E-13
	Rh-103m	3.90E-02	0.9970			1.78E+01	2.33E-02	6.30E-07	5.4E-18	6.00E-03	8.93E-16
Ru-106		3.68E+02				1.88E-03	8.17E+00	2.21E-04	6.9E-06	3.00E-06	2.31E+00
	Rh-106	3.45E-04	1.0000			2.01E+03	2.95E-05	7.97E-10	6.9E-06	NA[6]	
Ag-110m		2.50E+02				2.77E-03	2.67E+00	7.22E-05	4.4E-07	6.00E-06	7.36E-02
	Ag-110	2.85E-04	0.0133			2.43E+03			5.9E-09	NA[6]	
Te-129m		3.36E+01				2.06E-02	4.29E+01	1.16E-03	3.9E-20	7.00E-06	5.62E-15
	Te-129	4.83E-02	0.6500			1.44E+01			2.6E-20	4.00E-04	6.41E-17
Te-131m		1.25E+00				5.55E-01	4.85E+00	1.31E-04	0.0E+00	8.00E-06	0.00E+00
	Te-131	1.74E-02	0.2220			3.98E+01			0.0E+00	8.00E-05	0.00E+00

NAPS COL 2.0-24-A

Table 2.4-206 Results of Groundwater Transport Analysis Considering Radioactive Decay Only

Parent Radionuclide	Progeny in Chain	Half-life[1] (days)	Branching Fraction[1] d_{12}	d_{13}	d_{23}	Decay Rate[2] ($days^{-1}$)	Collection Tank Conc[3] (MBq/m^3)	Collection Tank Conc ($\mu Ci/cm^3$)	Ground Water Conc[4] ($\mu Ci/cm^3$)	ECL[5] ($\mu Ci/cm^3$)	Ground Water Conc / ECL
	I-131	8.04E+00		0.7780	1.0000	8.62E-02	6.89E+02	1.86E-02	2.8E-71	1.00E-06	2.78E-65
Te-132		3.26E+00				2.13E-01	1.21E+00	3.27E-05	5.9E-175	9.00E-06	6.56E-170
	I-132	9.58E-02	1.0000			7.24E+00	6.58E+01	1.78E-03	6.1E-175	1.00E-04	6.08E-171
I-133		8.67E-01				7.99E-01	5.51E+02	1.49E-02	0.0E+00	7.00E-06	0.00E+00
	Xe-133m	2.19E+00	0.0290			3.17E-01			3.5E-257	NA[6]	
	Xe-133	5.25E+00		0.9710	1.0000	1.32E-01			9.5E-109	NA[6]	
I-134		3.65E-02				1.90E+01	4.38E+01	1.18E-03	0.0E+00	4.00E-04	0.00E+00
I-135		2.75E-01				2.52E+00	2.19E+02	5.92E-03	0.0E+00	3.00E-05	0.00E+00
	Xe-135m	1.06E-02	0.1540			6.53E+01			0.0E+00	NA[6]	
	Xe-135	3.79E-01		0.8460	1.0000	1.83E+00			0.0E+00	NA[6]	
Cs-134		7.53E+02				9.21E-04	7.36E+01	1.99E-03	3.7E-04	9.00E-07	4.07E+02
Cs-136		1.31E+01				5.29E-02	7.25E+00	1.96E-04	1.1E-46	6.00E-06	1.87E-41
Cs-137		1.10E+04				6.30E-05	2.09E+02	5.65E-03	5.0E-03	1.00E-06	5.03E+03
	Ba-137m	1.77E-03	0.9460			3.91E+02	3.71E-03	1.00E-07	4.8E-03	NA[6]	
Cs-138		2.24E-02				3.09E+01	5.62E+00	1.52E-04	0.0E+00	4.00E-04	0.00E+00
Ba-140		1.27E+01				5.46E-02	1.75E+02	4.73E-03	1.3E-46	8.00E-06	1.58E-41
	La-140	1.68E+00	1.0000			4.13E-01	2.62E+01	7.08E-04	1.5E-46	9.00E-06	1.62E-41
Ce-141		3.25E+01				2.13E-02	2.97E+01	8.03E-04	7.6E-21	3.00E-05	2.52E-16
Ce-144		2.84E+02				2.44E-03	7.86E+00	2.12E-04	2.4E-06	3.00E-06	7.97E-01

NAPS COL 2.0-24-A

Table 2.4-206 Results of Groundwater Transport Analysis Considering Radioactive Decay Only

Parent Radionuclide	Progeny in Chain	Half-life[1] (days)	Branching Fraction[1] d12	d13	d23	Decay Rate[2] (days^{-1})	Collection Tank Conc[3] (MBq/m^3)	Collection Tank Conc (μCi/cm^3)	Ground Water Conc[4] (μCi/cm^3)	ECL[5] (μCi/cm^3)	Ground Water Conc / ECL
	Pr-144m	5.07E-03	0.0178			1.37E+02			4.3E-08	NA[6]	
	Pr-144	1.20E-02		0.9822	0.9990	5.78E+01	1.03E-03	2.78E-08	2.4E-06	6.00E-04	3.99E-03
W-187		9.96E-01				6.96E-01	1.15E+01	3.11E-04	0.0E+00	3.00E-05	0.00E+00
Np-239		2.36E+00				2.94E-01	7.17E+02	1.94E-02	6.5E-237	2.00E-05	3.24E-232
	Pu-239	8.79E+06	1.0000			7.89E-08			5.2E-09	2.00E-08	2.60E-01

1. Values from Table E.1, NUREG/CR-5512 (Reference 2.4-214) and ICRP Publication 38 (Reference 2.4-211) for Sr-92, Rh-106, Ag-110, Ba-137m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Pr-144m.
2. Values calculated from Equation 2.4.13-5.
3. Values from DCD Table 12.2-13a.
4. Values calculated from Equation 2.4.13-9, Equation 2.4.13-14, or Equation 2.4.13-19 depending on position in decay chain for a travel time of 5.03 years.
5. Values from 10 CFR 20, Appendix B, Table 2, Column 2.
6. ECL is not available.

NAPS COL 2.0-24-A

Table 2.4-207 Comparison of Site-Specific K_d Values Against NUREG/CR-6697 Derived Values

	K_d (cm^3/g)										
Sample	**Mn**	**Fe**	**Co**	**Ni**	**Zn**	**Sr**	**Ru**	**Ag**	**Cs**	**Ce**	**Pu**
B-949/R3	>8,145	>45,497	>15,765	>1,616	>5,110	68.5	>1,148	>31,091	>19,504	>10,422	8,680
B-951/R5	>12,196	>20,291	>18,778	>892	>4,217	60.2	>1,200	>12,729	6,863	>10,232	443
B-901/R20	>7,858	>5,146	2,364	>615	>2,411	14.8	>632	>12,792	387	>6,753	295
B-901/R22	5,499	>14,207	5,459	>811	>4,147	33	>988	>9,903	574	>7,073	351
B-901/S5	4.5	>13,456	6.5	40.6	11.8	3.9	>272	28.6	68	329.1	5.3
B-901/S8	>6,525	>5,646	>9,423	12.7	>7,190	166.4	>1,448	28.6	181	>9,572	34.3
B-904/S10	36.9	>12,489	58.3	342	136	3.6	>328	73.2	241	4,175	96.5
B-913/S9	12,492	>14,397	13,082	129	>5,901	14.5	>1,429	43.4	796	>10,149	177
B-913/S10	7,903	>6,505	5,711	162	>6,702	8.4	>1,080	6	141	>9,182	735
B-917/S12	8,046	>30,209	5,747	643	>5,511	7.6	>1,171	25.7	154	>8,831	305
B-917/S14	>10,470	>16,121	6,559	17.7	>4,563	6.6	>936	32.6	118.9	>6,893	209
B-917/S15	4,692	>4,504	3,991	53.3	>2,764	3.8	>524	16.6	64.9	>5,419	192
B-919/S8	>4,121	>40,524	3,840	387	>3,426	14.8	>1,007	232	378	>7,750	896
B-920/S11	>15,785	>19,392	8,768	>623	>7,905	25.5	>1,593	>482	379	>12,056	311
B-928/S7	3,801	>6,104	3,244	>424	>8,103	7.6	>1,212	>304	104	>11,468	528
B-929/S12	3,453	>19,967	5,331	45	>6,270	7.1	>1,264	2.5	104.9	>8,887	536
B-931/S11	3,988	>28,132	5,151	>369	>6,070	4.7	>1,149	44.4	67.5	>10,519	333
B-932/S6	9,013	>16,288	6,739	766	>5,684	11.2	>1,367	>12,665	159	10,449	2,488
B-951/S7	>21,374	>25,330	>20,653	>806	>6,991	26.8	>1,665	>12,716	3,406	>12,914	3,874
B-951/S9	6,143	>24,220	8,818	>658	>6,162	12.7	>1,472	>8,190	336	>13,194	3,603

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Table 2.4-207 Comparison of Site-Specific K_d Values Against NUREG/CR-6697 Derived Values

	K_d (cm^3/g)										
Sample	**Mn**	**Fe**	**Co**	**Ni**	**Zn**	**Sr**	**Ru**	**Ag**	**Cs**	**Ce**	**Pu**
Min =	4.5	4504	6.5	12.7	11.8	3.6	272	2.5	64.9	329.1	5.3
10% =	3111.4	5596.0	2133.4	38.3	2183.5	3.9	504.4	15.5	68.0	5294.6	90.3
25% =	4087.8	10993.0	3953.3	110.1	3966.8	7.0	975.0	28.6	115.4	7028.0	204.8
50% =	7191.5	16204.5	5729.0	405.5	5597.5	12.0	1160.0	152.6	211.0	9377.0	342.0
Max =	21374	45497	20653	1616	8103	166.4	1665	31091	19504	13194	8680
Mean =	7577.3	18421.3	7474.4	470.6	4963.7	25.1	1094.3	5070.3	1701.4	8813.4	1204.6
NUREG K_d =	8.37	6.81	9.19	65.3	3.63	2.08	28.75	14.71	22.51	138.99	84.59
Percentile[a] =	0.60%	< Min	0.20%	21.80%	< Min	< Min	< Min	9.50%	< Min	< Min	9.50%

a. Rank of NUREG K_d value as a percentage of the site-specific K_d data.

Table 2.4-208 Results of Groundwater Transport Analysis Considering Radioactive Decay and Adsorption

Parent Radionuclide	Progeny in Chain	Decay Rate[1] ($days^{-1}$)	Branching Fraction[2]	Initial Conc ($\mu Ci/cm^3$)	Literature Kd[3] m	Literature Kd[3] s	10% Kd (cm^3/g)	Retard Factor[4]	Ground Water Conc[5] ($\mu Ci/cm^3$)	ECL[6] ($\mu Ci/cm^3$)	Ground Water Conc / ECL
H-3		1.54E-04		2.63E-03				1.00	1.98E-03	1.00E-03	1.98E+00
Mn-54		2.21E-03		2.66E-03	5.06	2.29	8.37	62.25	2.30E-113	3.00E-05	7.68E-109
Fe-55		7.03E-04		8.32E-02	5.34	2.67	6.81	50.80	2.57E-30	1.00E-04	2.57E-26
Co-60		3.59E-04		1.69E-02	5.46	2.53	9.19	68.19	4.76E-22	3.00E-06	1.59E-16
Ni-63		1.97E-05		8.76E-05	6.05	1.46	65.30	478.58	2.50E-12	1.00E-04	2.50E-08
Zn-65		2.84E-03		7.16E-02	6.98	4.44	3.63	27.57	2.16E-64	5.00E-06	4.32E-59
Sr-90		6.54E-05		6.03E-04	3.45	2.12	2.08	16.22	8.57E-05	5.00E-07	1.71E+02
	Y-90	2.60E-01	1.0000	1.88E-05	6.84	3.22	15.08	111.30	8.57E-05	7.00E-06	1.22E+01
Ru-106		1.88E-03		2.21E-04	7.37	3.13	28.75	211.30	0.00E+00	3.00E-06	0.00E+00
Ag-110m		2.77E-03		7.22E-05	5.38	2.10	14.71	108.61	2.82E-245	6.00E-06	4.70E-240
Cs-134		9.21E-04		1.99E-03	6.10	2.33	22.51	165.64	3.73E-125	9.00E-07	4.14E-119
Cs-137		6.30E-05		5.65E-03	6.10	2.33	22.51	165.64	2.63E-11	1.00E-06	2.63E-05
Ce-144		2.44E-03		2.12E-04	7.60	2.08	138.99	1017.54	0.00E+00	3.00E-06	0.00E+00
Np-239		2.94E-01		1.94E-02	2.84	2.25	0.96	8.00	0.00E+00	2.00E-05	0.00E+00
	Pu-239	7.89E-08	1.0000	0.00E+00	6.86	1.89	84.59	619.72	3.68E-07	2.00E-08	1.84E+01

1. Values calculated from Equation 2.4.13-5.
2. Values from Table E.1, NUREG/CR-5512 (Reference 2.4-214).
3. Mean and standard deviation from NUREG/CR-6697, Attachment C, Table 3.9-1 (Reference 2.4-215); Sc values used as surrogates for Y.
4. Values calculated from Equation 2.4.13-3.
5. Values calculated from Equation 2.4.13-9 for parent and Equation 2.4.13-14 for progeny.
6. Values from 10 CFR 20, Appendix B, Table 2, Column 2.

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Table 2.4-209 Results of Groundwater Transport Analysis Considering Radioactive Decay, Adsorption, and Dilution

Radionuclide	Groundwater Concentration[1] (μCi/cm^3)	Surface Water Concentration[2] (μCi/cm^3)	ECL[3]	Surface Water Concentration / ECL
H-3	1.98E-03	5.08E-07	1.00E-03	5.08E-04
Sr-90	8.57E-05	2.20E-08	5.00E-07	4.40E-02
Y-90	8.57E-05	2.20E-08	7.00E-06	3.14E-03
Pu-239	3.68E-07	9.45E-11	2.00E-08	4.72E-03

1. Values from Table 2.4-208.
2. Surface water concentration = groundwater concentration * dilution factor of 2.56×10^{-4}.
3. Values from 10 CFR 20, Appendix B, Table 2, Column 2.

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Table 2.4-210 Compliance with 10 CFR Part 20 for an Accidental Release of Radioactive Liquid Effluent in Groundwater

Parent Radionuclide	Progeny in Chain	Concentration/ECL Decay[1]	Decay and Adsorption[2]	Decay, Adsorption, and Dilution[3]	Minimum
H-3		1.98E+00	1.98E+00	5.08E-04	5.08E-04
Na-24		0.00E+00			0.00E+00
P-32		1.20E-37			1.20E-37
Cr-51		1.49E-18			1.49E-18
Mn-54		1.51E+00	7.68E-109		7.68E-109
Mn-56		0.00E+00			0.00E+00
Fe-55		2.29E+02	2.57E-26		2.57E-26
Fe-59		3.79E-11			3.79E-11
Co-58		3.63E-06			3.63E-06
Co-60		2.91E+03	1.59E-16		1.59E-16
Ni-63		8.44E-01	2.50E-08		2.50E-08
Cu-64		0.00E+00			0.00E+00
Zn-65		7.73E+01	4.32E-59		4.32E-59
Rb-89		0.00E+00			0.00E+00
	Sr-89	5.33E-09			5.33E-09
Sr-90		1.07E+03	1.71E+02	4.40E-02	4.40E-02
	Y-90	7.64E+01	1.22E+01	3.14E-03	3.14E-03
Sr-91		0.00E+00			0.00E+00
	Y-91m	0.00E+00			0.00E+00
	Y-91	7.42E-08			7.42E-08
Sr-92		0.00E+00			0.00E+00
Y-92		0.00E+00			0.00E+00
Y-93		0.00E+00			0.00E+00
Zr-95		4.09E-08			4.09E-08
	Nb-95m	2.02E-10			2.02E-10
	Nb-95	6.06E-08			6.06E-08

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Table 2.4-210 Compliance with 10 CFR Part 20 for an Accidental Release of Radioactive Liquid Effluent in Groundwater

		Concentration/ECL			
Parent Radionuclide	**Progeny in Chain**	**Decay[1]**	**Decay and Adsorption[2]**	**Decay, Adsorption, and Dilution[3]**	**Minimum**
Mo-99		1.67E-199			1.67E-199
	Tc-99m	3.23E-201			3.23E-201
Ru-103		1.79E-13			1.79E-13
	Rh-103m	8.93E-16			8.93E-16
Ru-106		2.31E+00	0.00E+00		0.00E+00
	Rh-106	0.00E+00			0.00E+00
Ag-110m		7.36E-02	4.70E-240		4.70E-240
	Ag-110	0.00E+00			0.00E+00
Te-129m		5.62E-15			5.62E-15
	Te-129	6.41E-17			6.41E-17
Te-131m		0.00E+00			0.00E+00
	Te-131	0.00E+00			0.00E+00
	I-131	2.78E-65			2.78E-65
Te-132		6.56E-170			6.56E-170
	I-132	6.08E-171			6.08E-171
I-133		0.00E+00			0.00E+00
	Xe-133m	0.00E+00			0.00E+00
	Xe-133	0.00E+00			0.00E+00
I-134		0.00E+00			0.00E+00
I-135		0.00E+00			0.00E+00
	Xe-135m	0.00E+00			0.00E+00
	Xe-135	0.00E+00			0.00E+00
Cs-134		4.07E+02	4.14E-119		4.14E-119
Cs-136		1.87E-41			1.87E-41
Cs-137		5.03E+03	2.63E-05		2.63E-05
	Ba-137m	0.00E+00			0.00E+00
Cs-138		0.00E+00			0.00E+00

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Table 2.4-210 Compliance with 10 CFR Part 20 for an Accidental Release of Radioactive Liquid Effluent in Groundwater

Parent Radionuclide	Progeny in Chain	Concentration/ECL: Decay[1]	Decay and Adsorption[2]	Decay, Adsorption, and Dilution[3]	Minimum
Ba-140		1.58E-41			1.58E-41
	La-140	1.62E-41			1.62E-41
Ce-141		2.52E-16			2.52E-16
Ce-144		7.97E-01	0.00E+00		0.00E+00
	Pr-144m	0.00E+00			0.00E+00
	Pr-144	3.99E-03			3.99E-03
W-187		0.00E+00			0.00E+00
Np-239		3.24E-232	0.00E+00		0.00E+00
	Pu-239	2.60E-01	1.84E+01	4.72E-03	4.72E-03
				Sum of Fractions =	5.64E-02

1. Table 2.4-206
2. Table 2.4-208
3. Table 2.4-209
4. No ECLs are published for Rh-106, Ag-110, Xe-133m, Xe-133, Xe-135m, Xe-135, Ba-137m, and Pr-144m. However, their half-lives are short (on the order of days or less) and they decay to near-zero concentrations. Their ratios have been taken as zero.

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Table 2.4-211 Dilution Factors for Various Plant Operating Scenarios

Scenario	Q_{CST} (ft^3/s)	Q_{3MU} (ft^3/s)	Q_{3BD} (ft^3/s)	Q_{12} (ft^3/s)	Q_{DC} (ft^3/s)	N	C_{CST} ($\mu Ci/cm^3$)	C_{3MU} ($\mu Ci/cm^3$)	C_{3BD} ($\mu Ci/cm^3$)	C_{DC} ($\mu Ci/cm^3$)
1	0.60	34	9	4246	4255	4	1.00E+00	1.76E-02	7.06E-02	1.49E-04
2	0.60	34	9	531	540	4	1.00E+00	1.76E-02	7.06E-02	1.18E-03
3	0.60	0	0	4246	4246	-	1.00E+00	-	-	1.41E-04
4	0.60	0	0	531	531	-	1.00E+00	-	-	1.13E-03

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Table 2.4-212 Compliance with 10 CFR 20 for an Accidental Release of Radioactive Liquid Effluent in Surface Water

		CST		Surface Water	
Radionuclide	ECL (μCi/cm^3)	Conc (MBq/m^3)	Conc (μCi/cm^3)	Conc (μCi/cm^3)	Conc / ECL
H-3	1.00E-03	3.7E+02	1.0E-02	1.2E-05	1.2E-02
Na-24	5.00E-05	3.2E-02	8.6E-07	1.0E-09	2.0E-05
P-32	9.00E-06	6.6E-04	1.8E-08	2.1E-11	2.3E-06
Cr-51	5.00E-04	5.0E-02	1.4E-06	1.6E-09	3.2E-06
Mn-54	3.00E-05	5.8E-04	1.6E-08	1.8E-11	6.2E-07
Mn-56	7.00E-05	3.8E-01	1.0E-05	1.2E-08	1.7E-04
Fe-55	1.00E-04	1.7E-02	4.6E-07	5.4E-10	5.4E-06
Fe-59	1.00E-05	5.0E-04	1.4E-08	1.6E-11	1.6E-06
Co-58	2.00E-05	1.7E-03	4.6E-08	5.4E-11	2.7E-06
Co-60	3.00E-06	3.3E-03	8.9E-08	1.1E-10	3.5E-05
Ni-63	1.00E-04	1.7E-05	4.6E-10	5.4E-13	5.4E-09
Cu-64	2.00E-04	4.8E-02	1.3E-06	1.5E-09	7.7E-06
Zn-65	5.00E-06	1.7E-02	4.6E-07	5.4E-10	1.1E-04
Rb-89	9.00E-04	3.5E-01	9.5E-06	1.1E-08	1.2E-05
Sr-89	8.00E-06	1.4E-01	3.8E-06	4.5E-09	5.6E-04
Sr-90	5.00E-07	2.2E-02	5.9E-07	7.0E-10	1.4E-03
Y-90	7.00E-06	4.0E-04	1.1E-08	1.3E-11	1.8E-06
Sr-91	2.00E-05	6.4E-02	1.7E-06	2.0E-09	1.0E-04
Y-91	8.00E-06	6.6E-04	1.8E-08	2.1E-11	2.6E-06
Sr-92	4.00E-05	1.5E-01	4.1E-06	4.8E-09	1.2E-04
Y-92	4.00E-05	9.3E-02	2.5E-06	3.0E-09	7.4E-05
Y-93	2.00E-05	6.4E-02	1.7E-06	2.0E-09	1.0E-04
Zr-95	2.00E-05	1.3E-04	3.5E-09	4.1E-12	2.1E-07
Nb-95	3.00E-05	1.3E-04	3.5E-09	4.1E-12	1.4E-07
Mo-99	2.00E-05	1.2E-01	3.2E-06	3.8E-09	1.9E-04
Tc-99m	1.00E-03	3.3E-02	8.9E-07	1.1E-09	1.1E-06
Ru-103	3.00E-05	3.3E-04	8.9E-09	1.1E-11	3.5E-07

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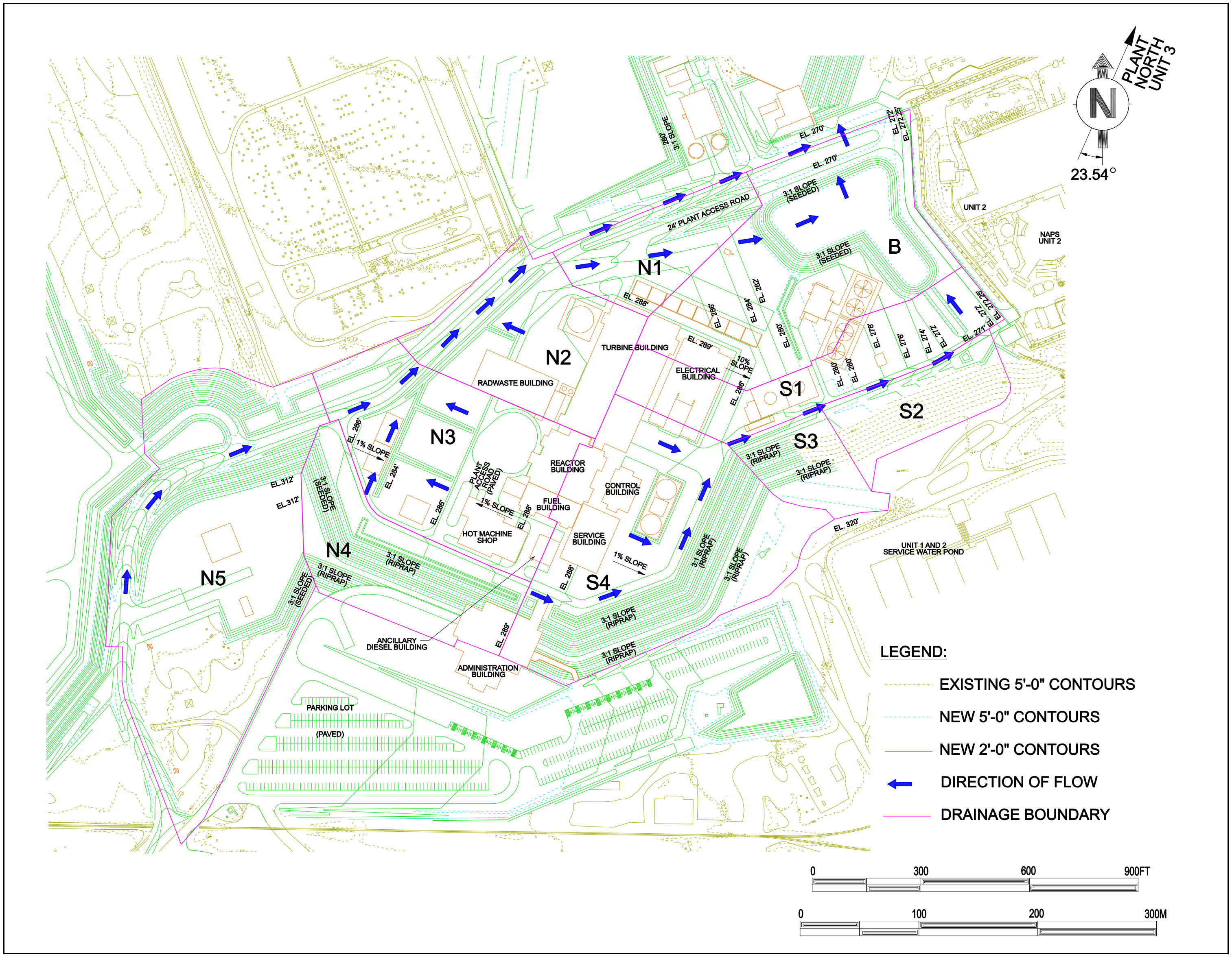
Table 2.4-212 Compliance with 10 CFR 20 for an Accidental Release of Radioactive Liquid Effluent in Surface Water

		CST		Surface Water	
Radionuclide	ECL (μCi/cm^3)	Conc (MBq/m^3)	Conc (μCi/cm^3)	Conc (μCi/cm^3)	Conc / ECL
Rh-103m	6.00E-03	3.3E-04	8.9E-09	1.1E-11	1.8E-09
Ru-106	3.00E-06	5.0E-05	1.4E-09	1.6E-12	5.3E-07
Rh-106	None	5.0E-05	1.4E-09	1.6E-12	
Ag-110m	6.00E-06	1.7E-05	4.6E-10	5.4E-13	9.0E-08
Te-129m	7.00E-06	4.1E-02	1.1E-06	1.3E-09	1.9E-04
Te-131m	8.00E-06	1.6E-03	4.3E-08	5.1E-11	6.4E-06
I-131	1.00E-06	7.9E-01	2.1E-05	2.5E-08	2.5E-02
Te-132	9.00E-06	7.6E-04	2.1E-08	2.4E-11	2.7E-06
I-132	1.00E-04	7.4E+00	2.0E-04	2.4E-07	2.4E-03
I-133	7.00E-06	5.3E+00	1.4E-04	1.7E-07	2.4E-02
I-134	4.00E-04	1.4E+01	3.8E-04	4.5E-07	1.1E-03
I-135	3.00E-05	7.6E+00	2.1E-04	2.4E-07	8.1E-03
Cs-134	9.00E-07	7.3E-01	2.0E-05	2.3E-08	2.6E-02
Cs-136	6.00E-06	6.5E-02	1.8E-06	2.1E-09	3.5E-04
Cs-137	1.00E-06	2.1E+00	5.7E-05	6.7E-08	6.7E-02
Ba-137m	None	1.2E-03	3.2E-08	3.8E-11	
Cs-138	4.00E-04	7.0E-01	1.9E-05	2.2E-08	5.6E-05
Ba-140	8.00E-06	1.6E-01	4.3E-06	5.1E-09	6.4E-04
La-140	9.00E-06	6.6E-03	1.8E-07	2.1E-10	2.3E-05
Ce-141	3.00E-05	5.0E-04	1.4E-08	1.6E-11	5.3E-07
Ce-144	3.00E-06	5.0E-05	1.4E-09	1.6E-12	5.3E-07
Pr-144	6.00E-04	5.0E-05	1.4E-09	1.6E-12	2.7E-09
W-187	3.00E-05	4.9E-03	1.3E-07	1.6E-10	5.2E-06
Np-239	2.00E-05	3.8E-01	1.0E-05	1.2E-08	6.1E-04
				Sum =	1.7E-01

Figure 2.4-201 Site Layout and Sub-Basin Drainage Areas



NAPS COL 2.0-13-A **Figure 2.4-202 Unit 3 Site PMP Duration- Intensity Curve**

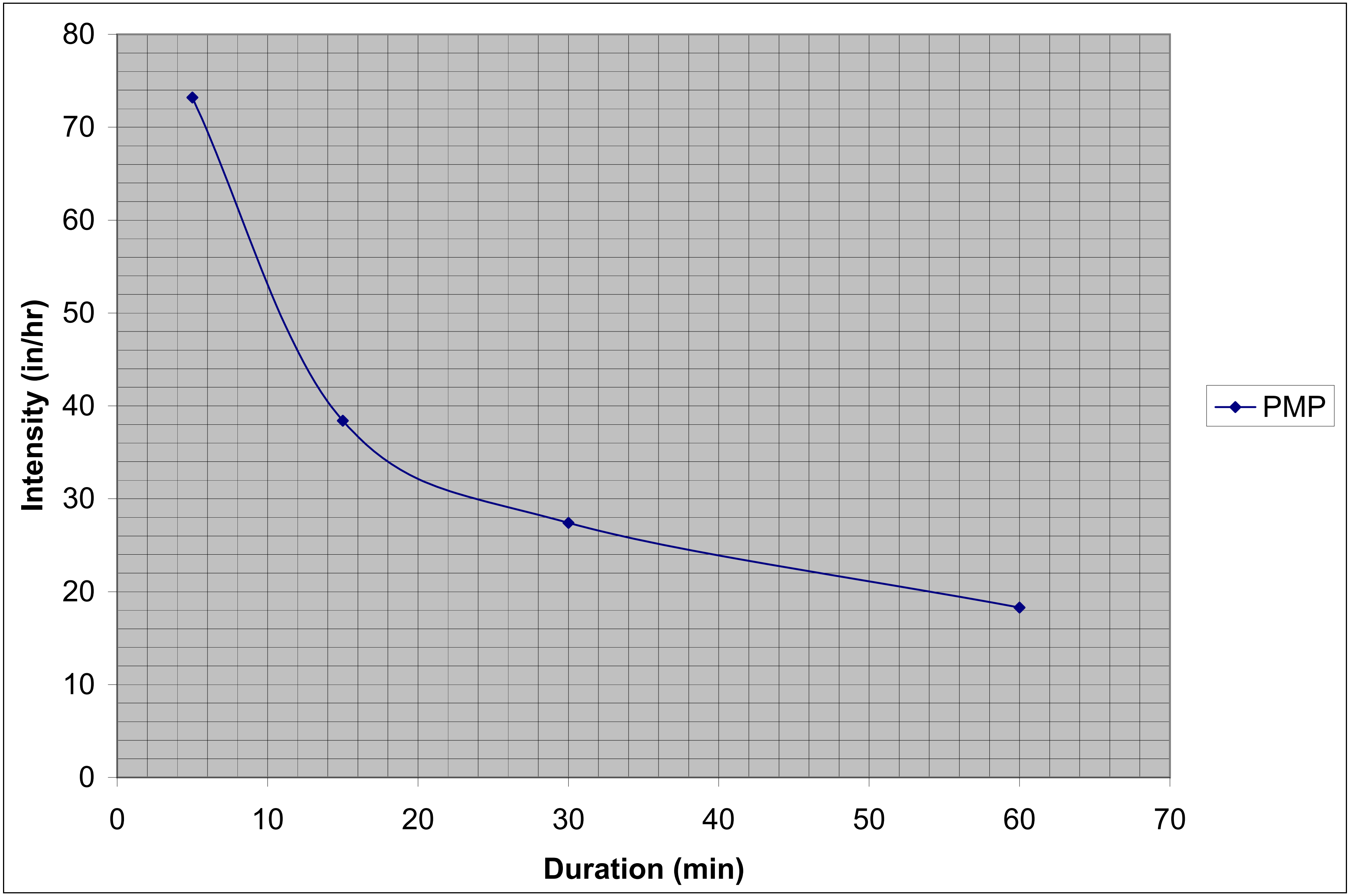


Figure 2.4-203 Cross-Section Locations

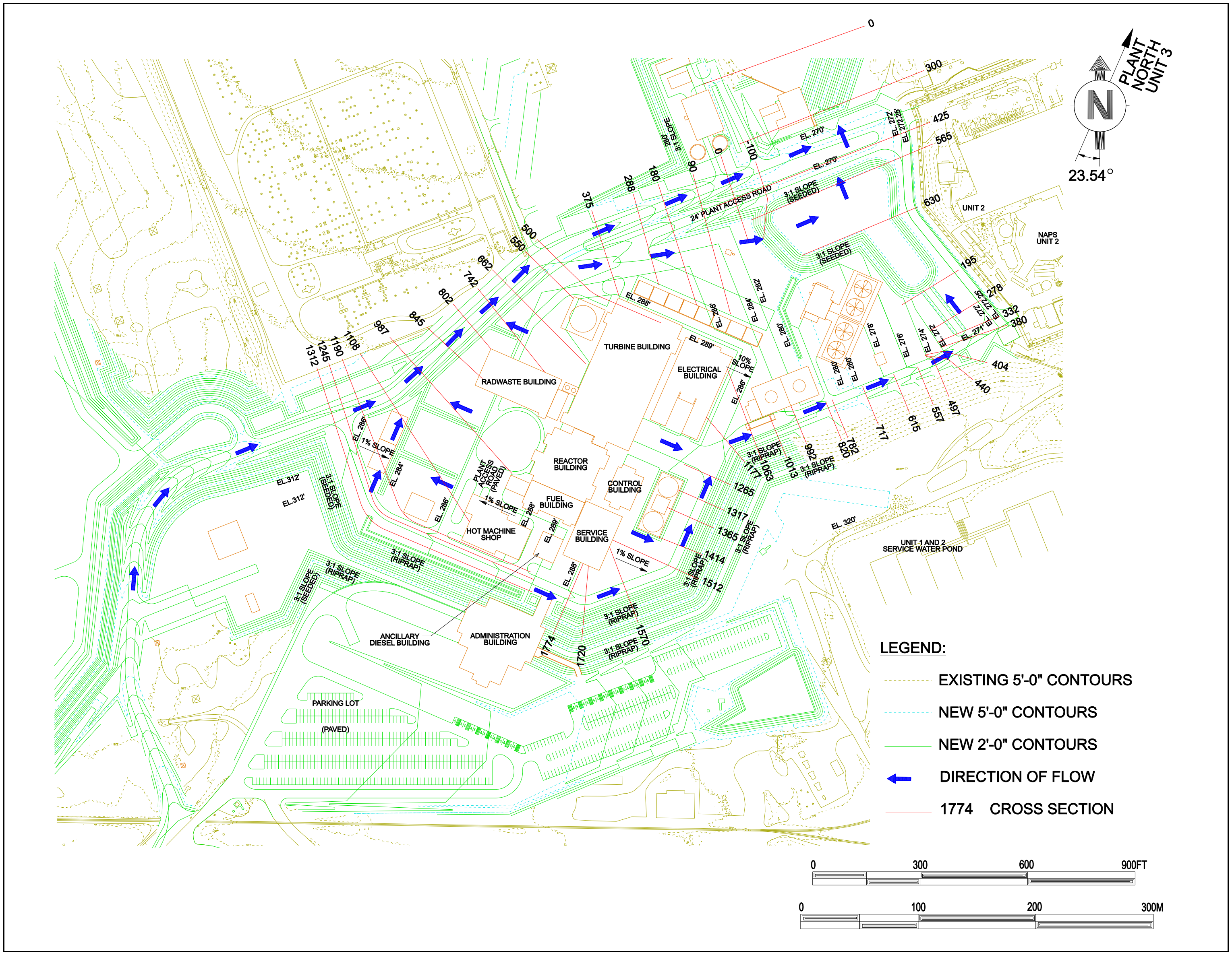
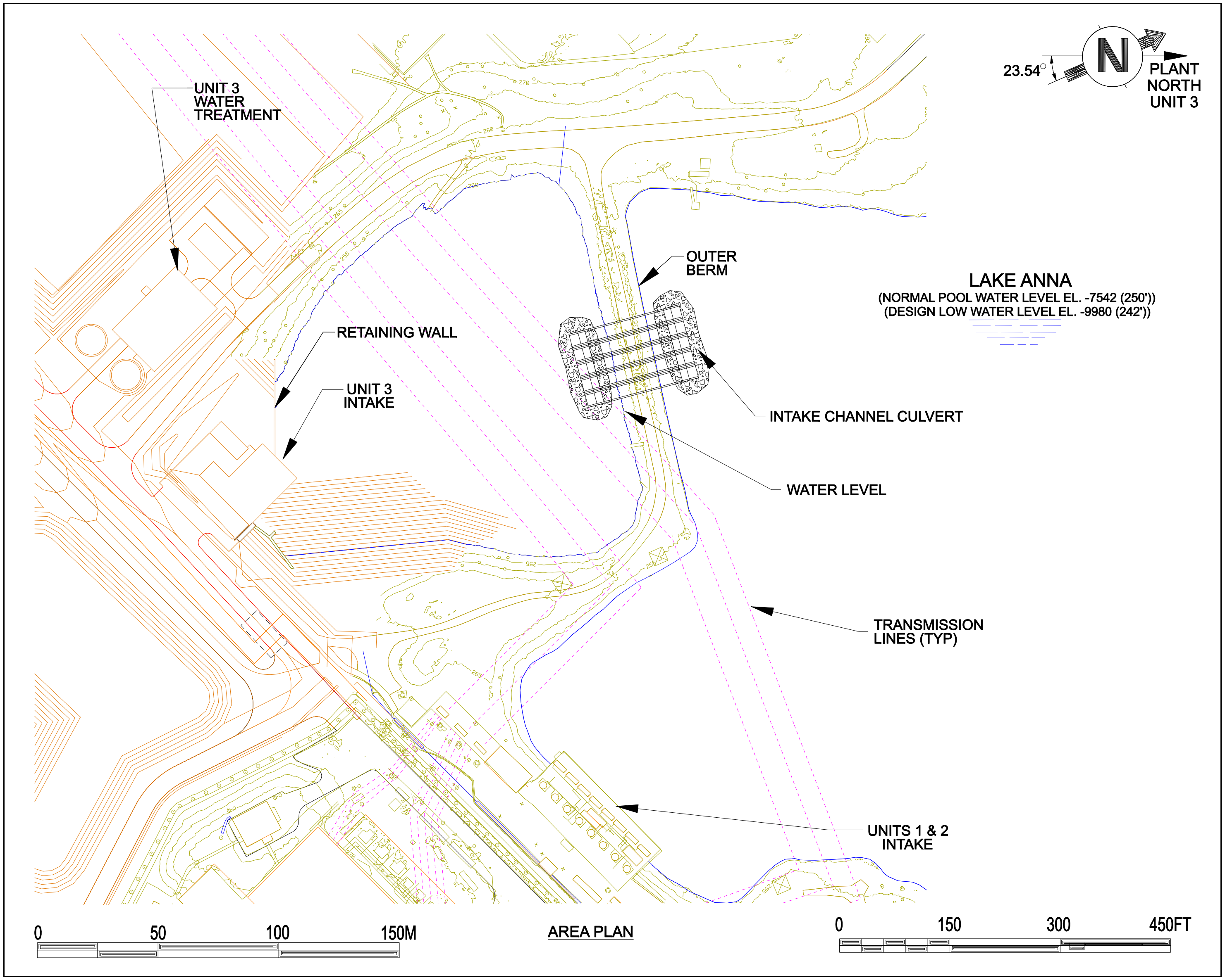


Figure 2.4-204 Unit 3 Make-up Water Intake Location



NAPS COL 2.0-23-A **Figure 2.4-205 Groundwater Level Hydrographs**

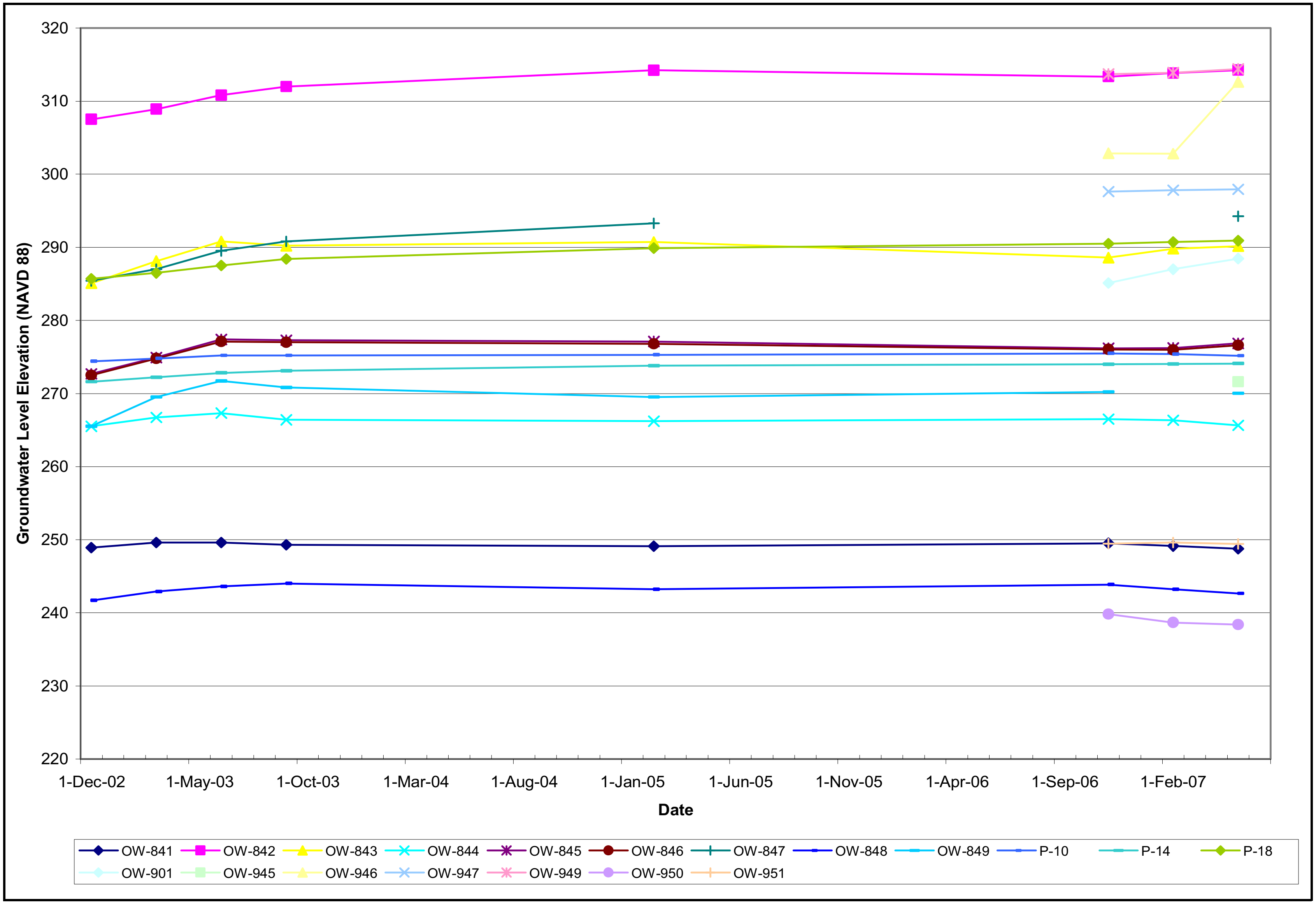


Figure 2.4-206 Observation Well Location Plan

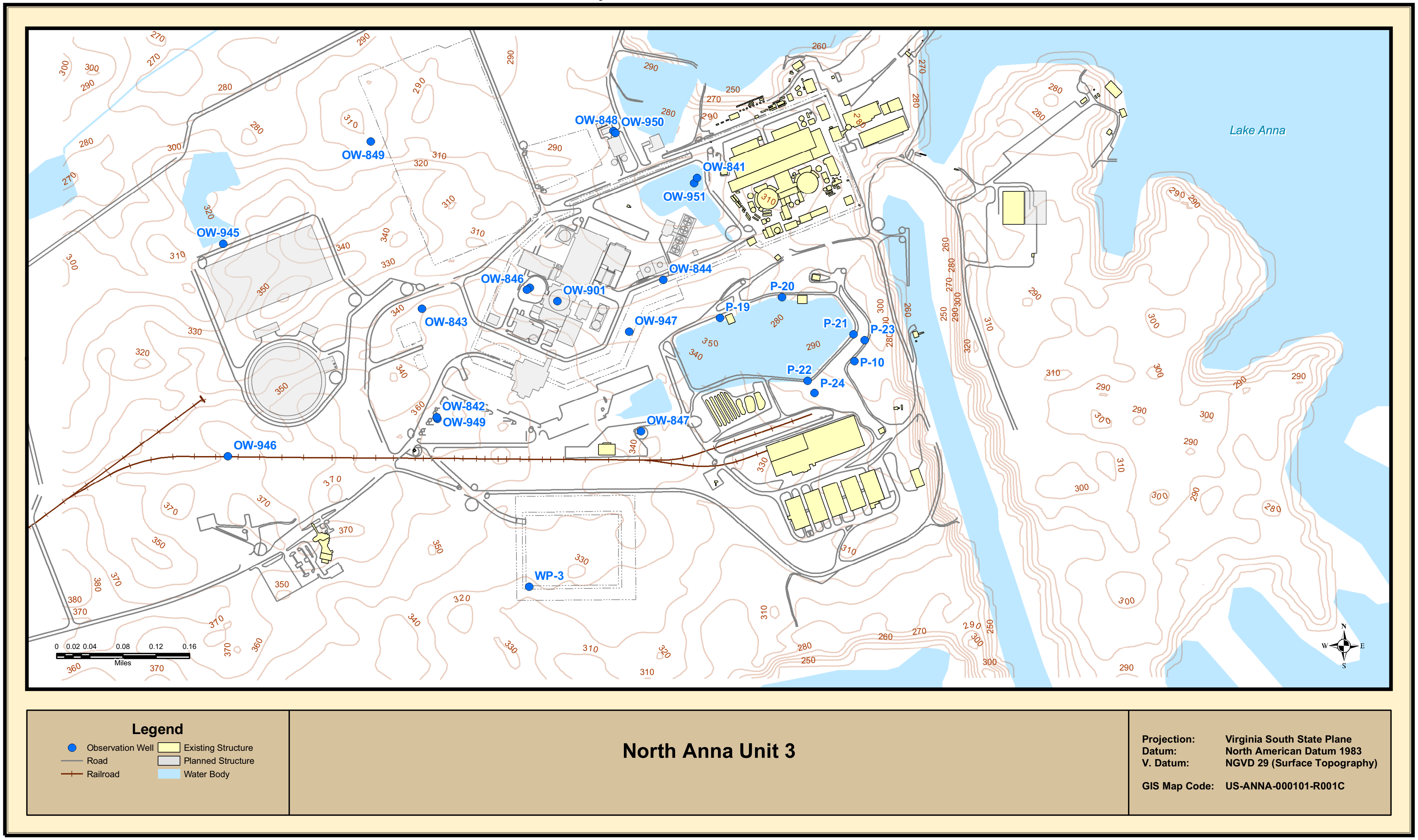


Figure 2.4-207 Piezometric Head Contour Map: December 2002

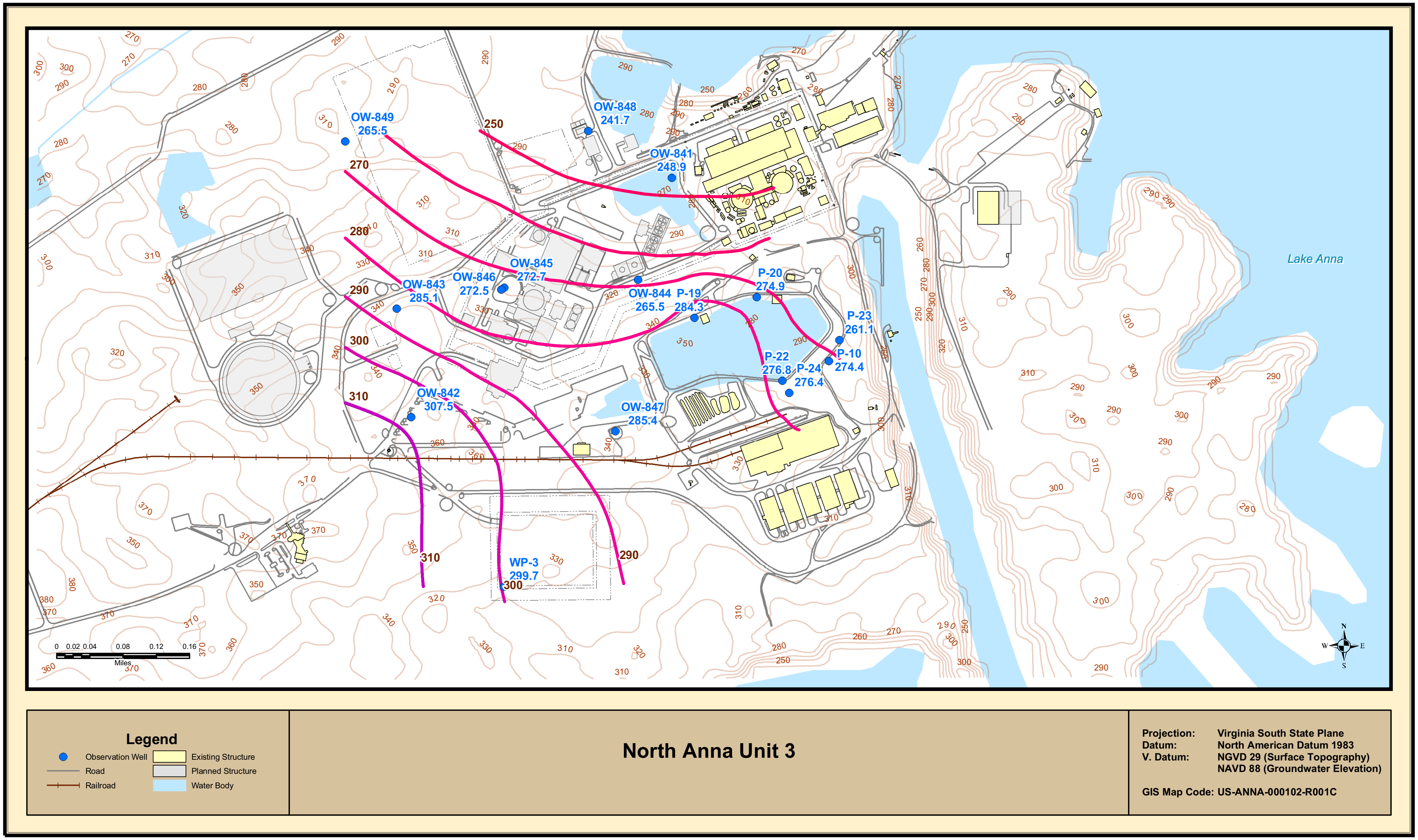
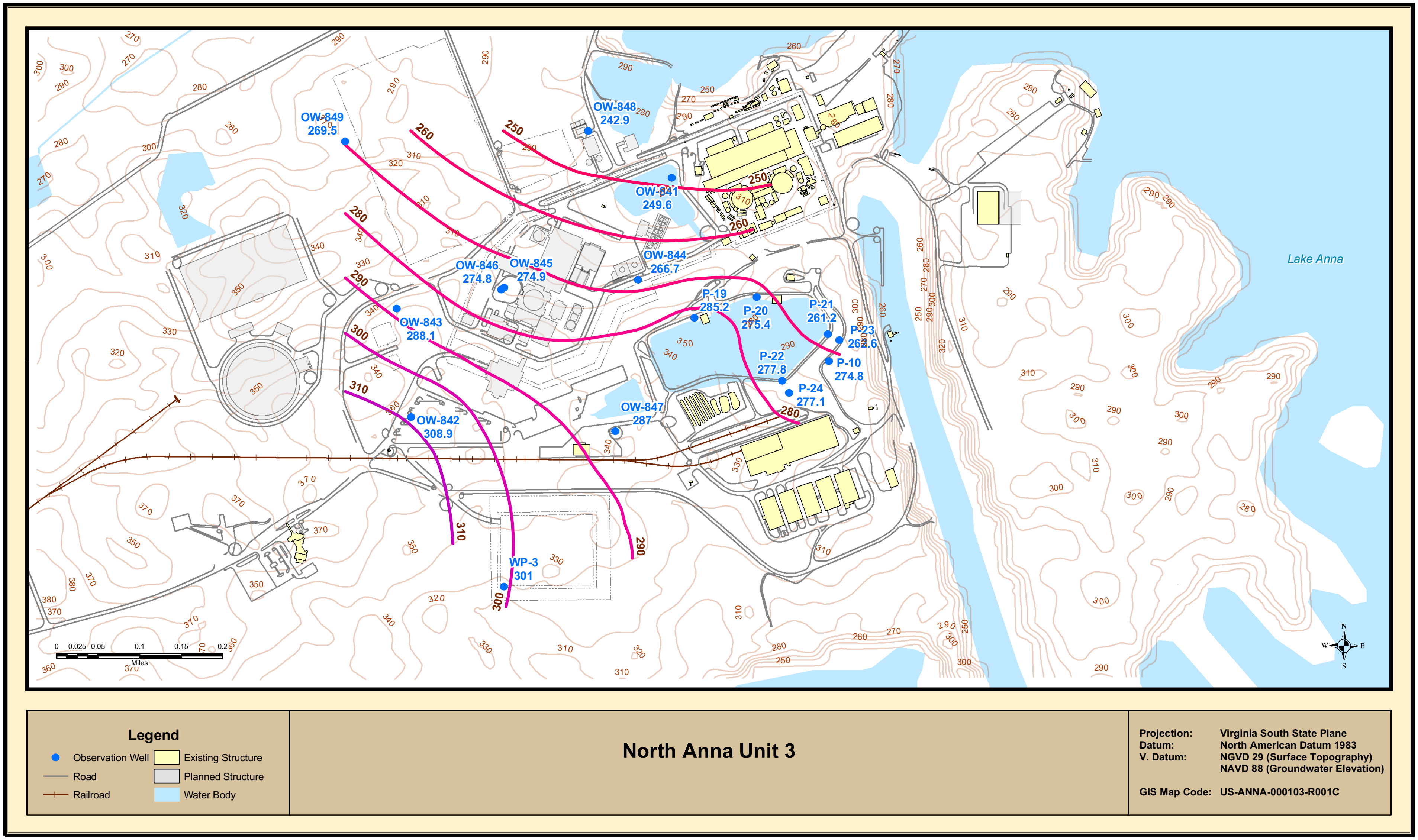


Figure 2.4-208 Piezometric Head Contour Map: March 2003



NAPS COL 2.0-23-A **Figure 2.4-209 Piezometric Head Contour Map: June 2003**

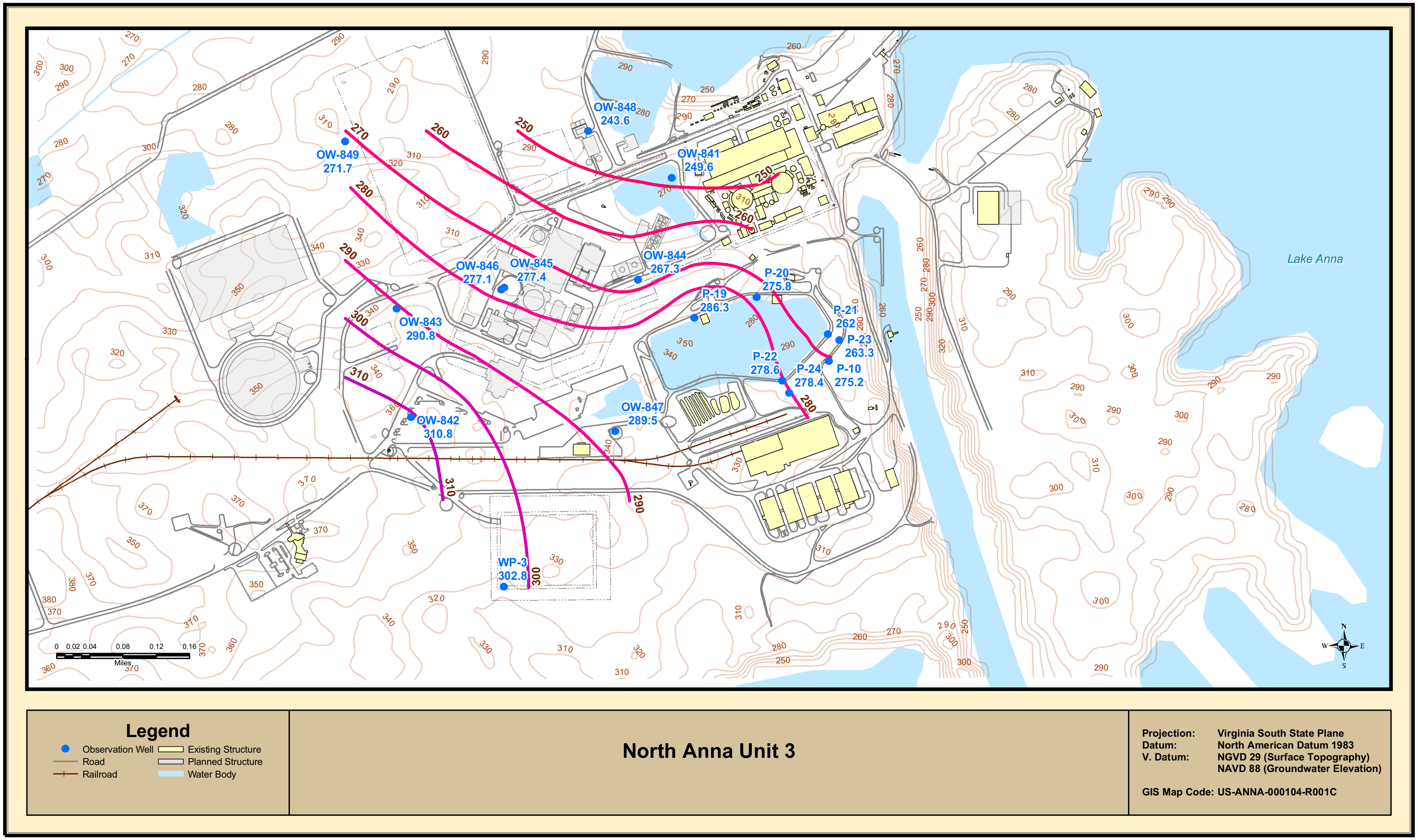


Figure 2.4-210 Piezometric Head Contour Map: September 2003

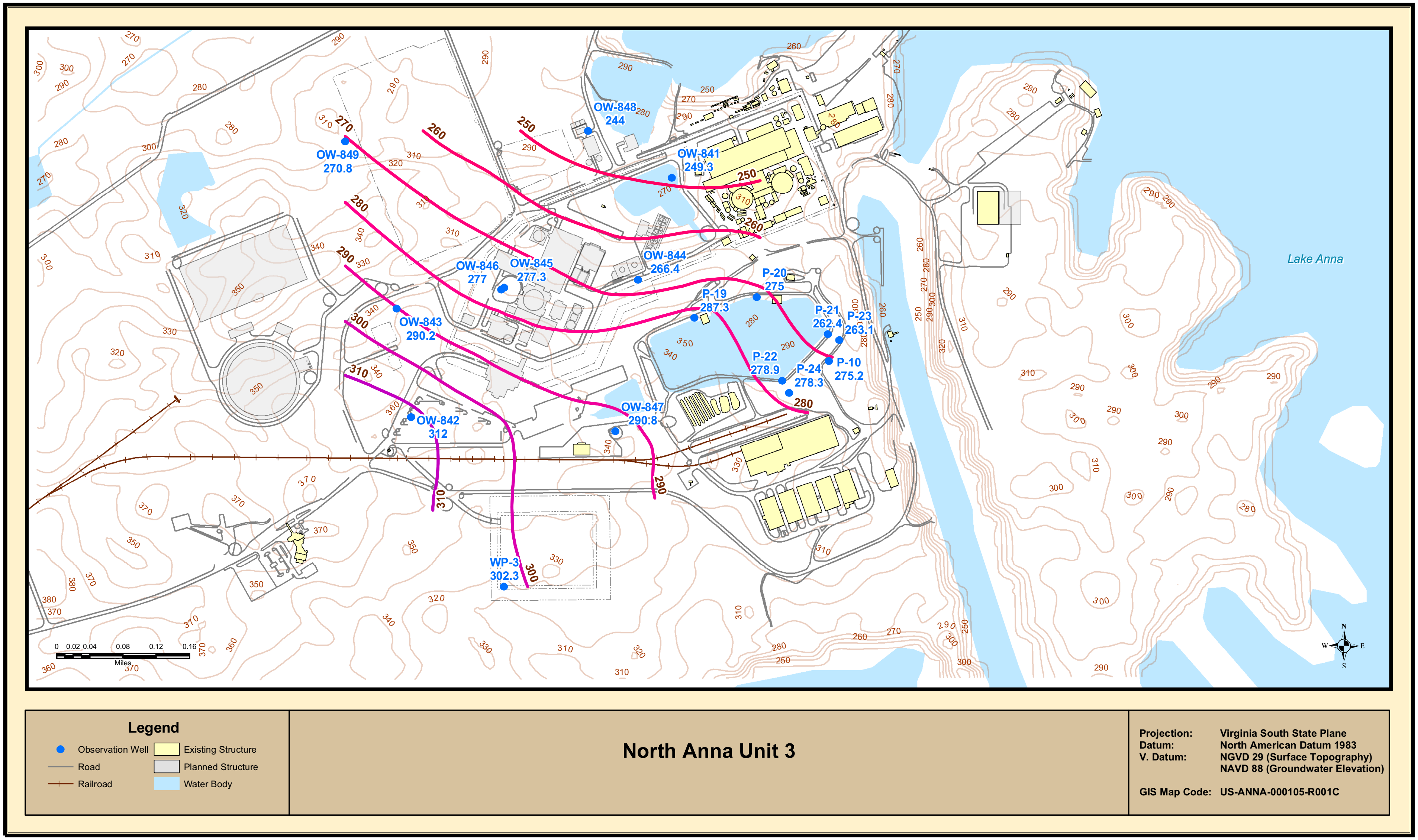


Figure 2.4-211 Piezometric Head Contour Map: February 2005

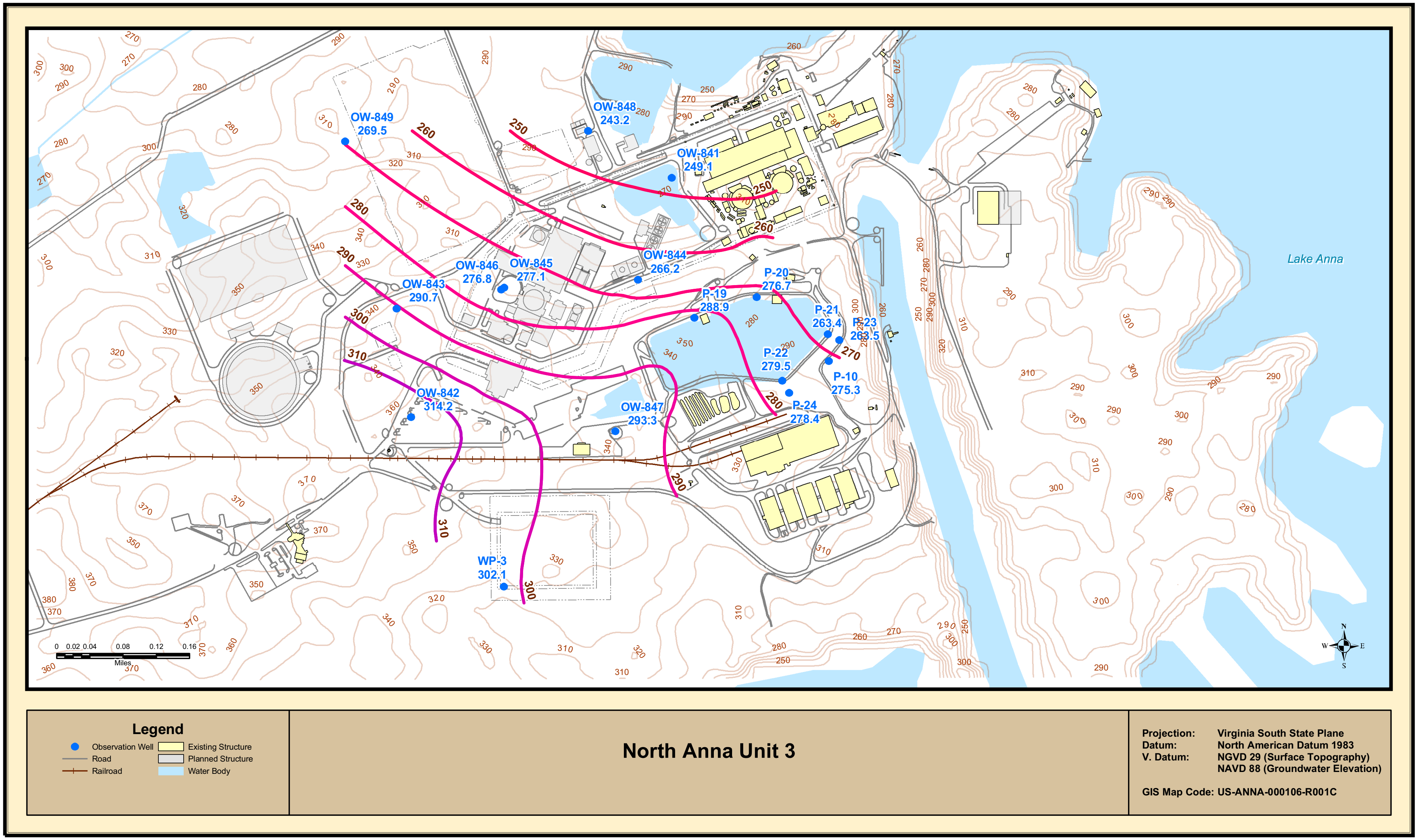


Figure 2.4-212 Piezometric Head Contour Map: November 2006

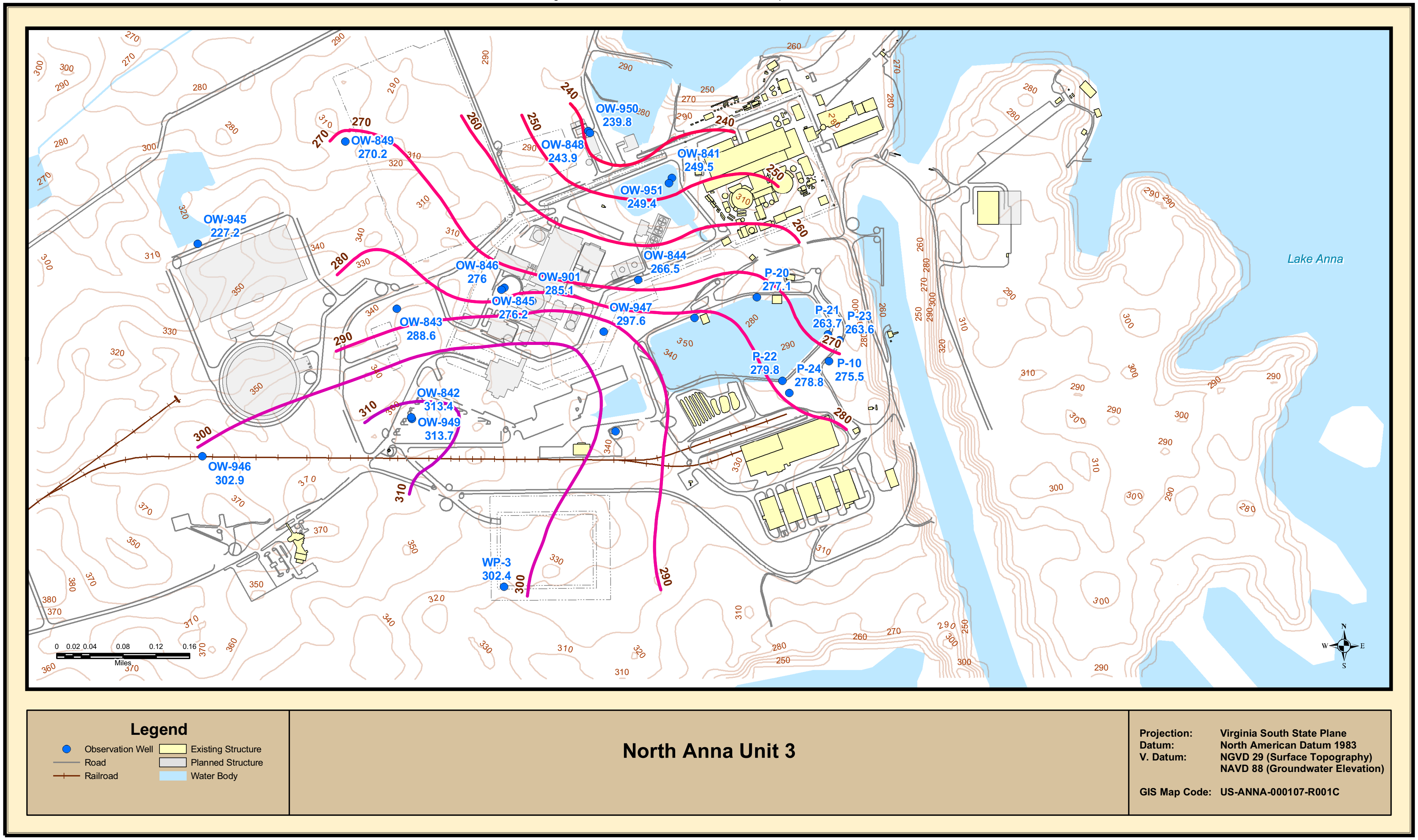


Figure 2.4-213 Piezometric Head Contour Map: February 2007

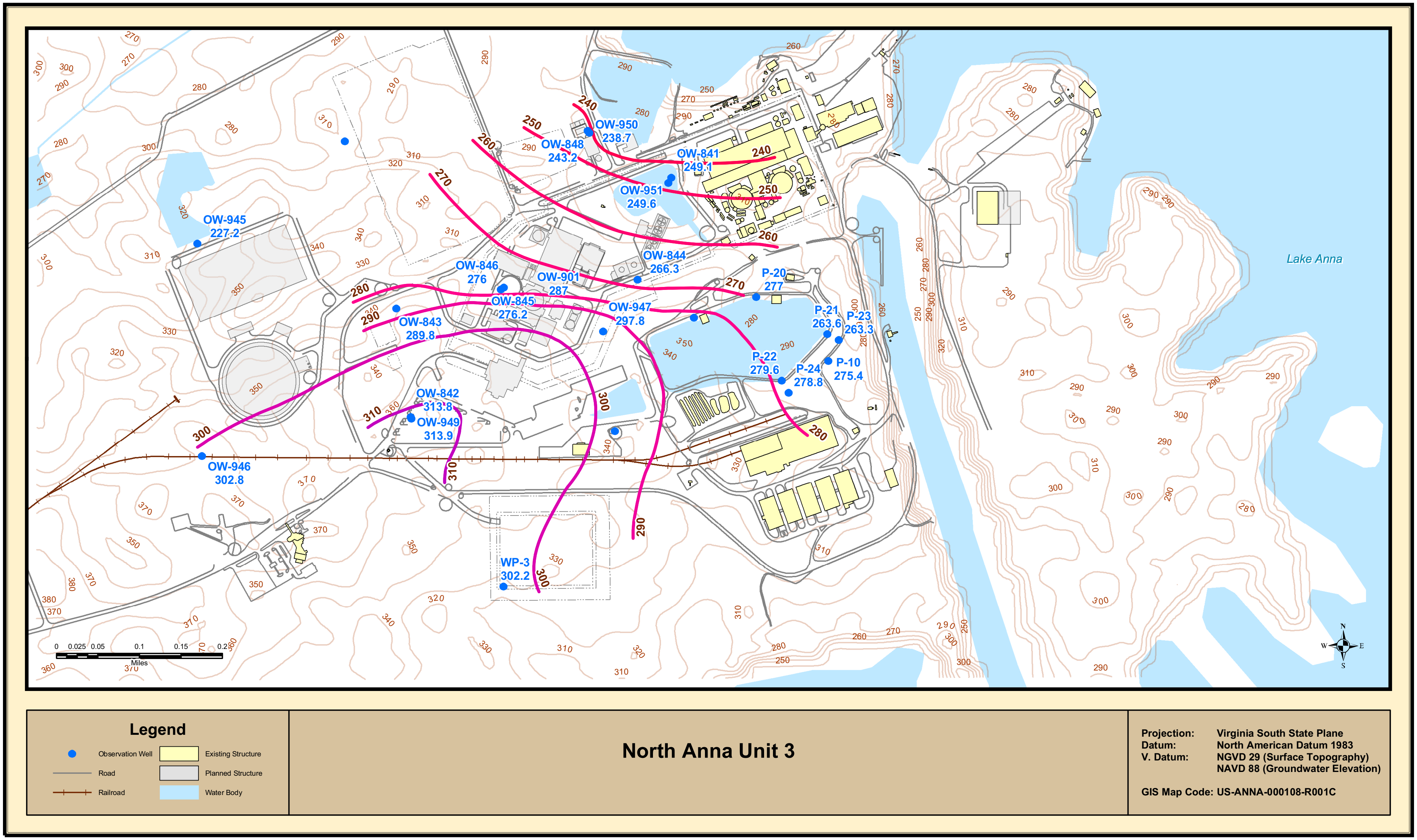
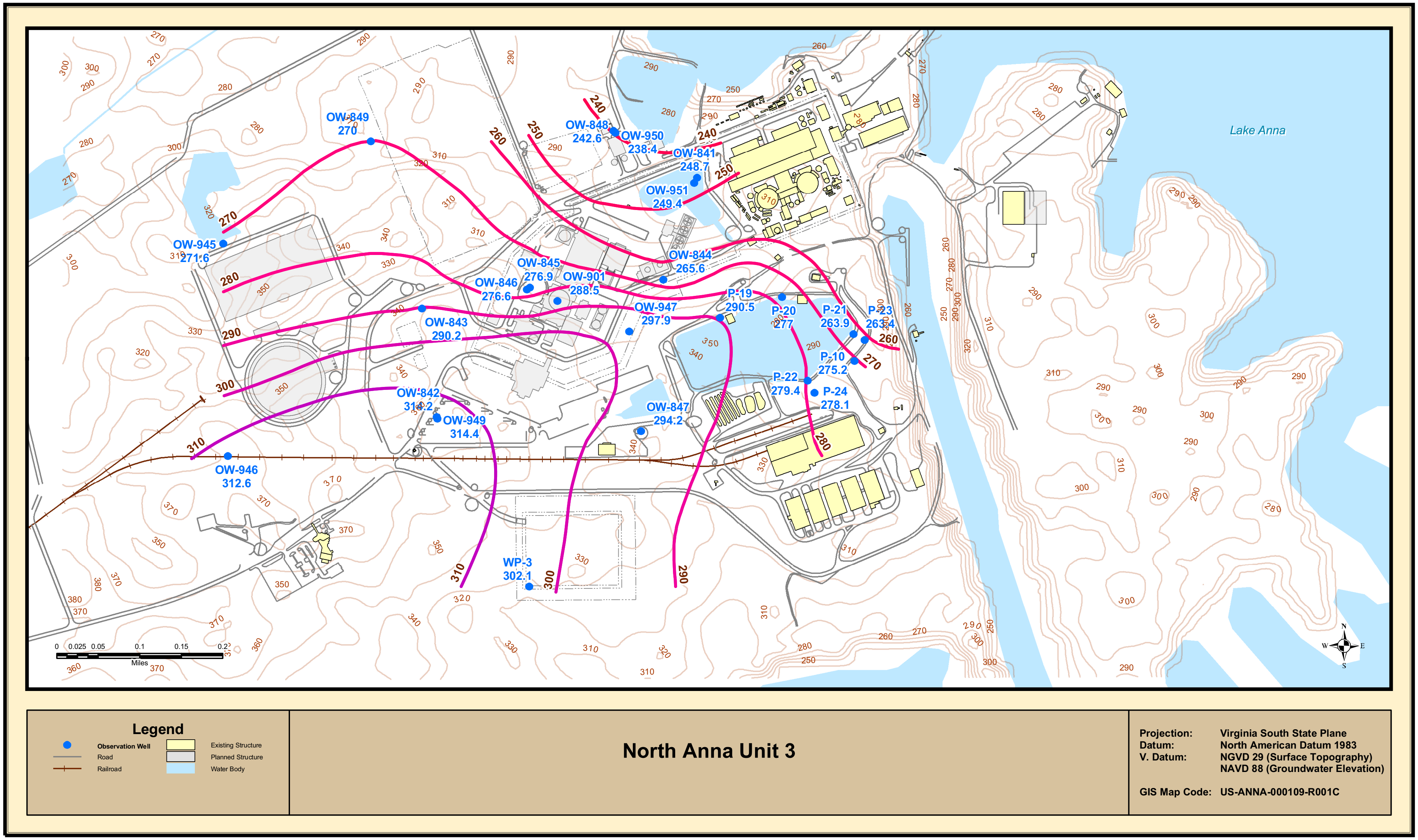


Figure 2.4-214 Piezometric Head Contour Map: May 2007



NAPS COL 2.0-23-A **Figure 2.4-215 Water Supply Well location Plan**

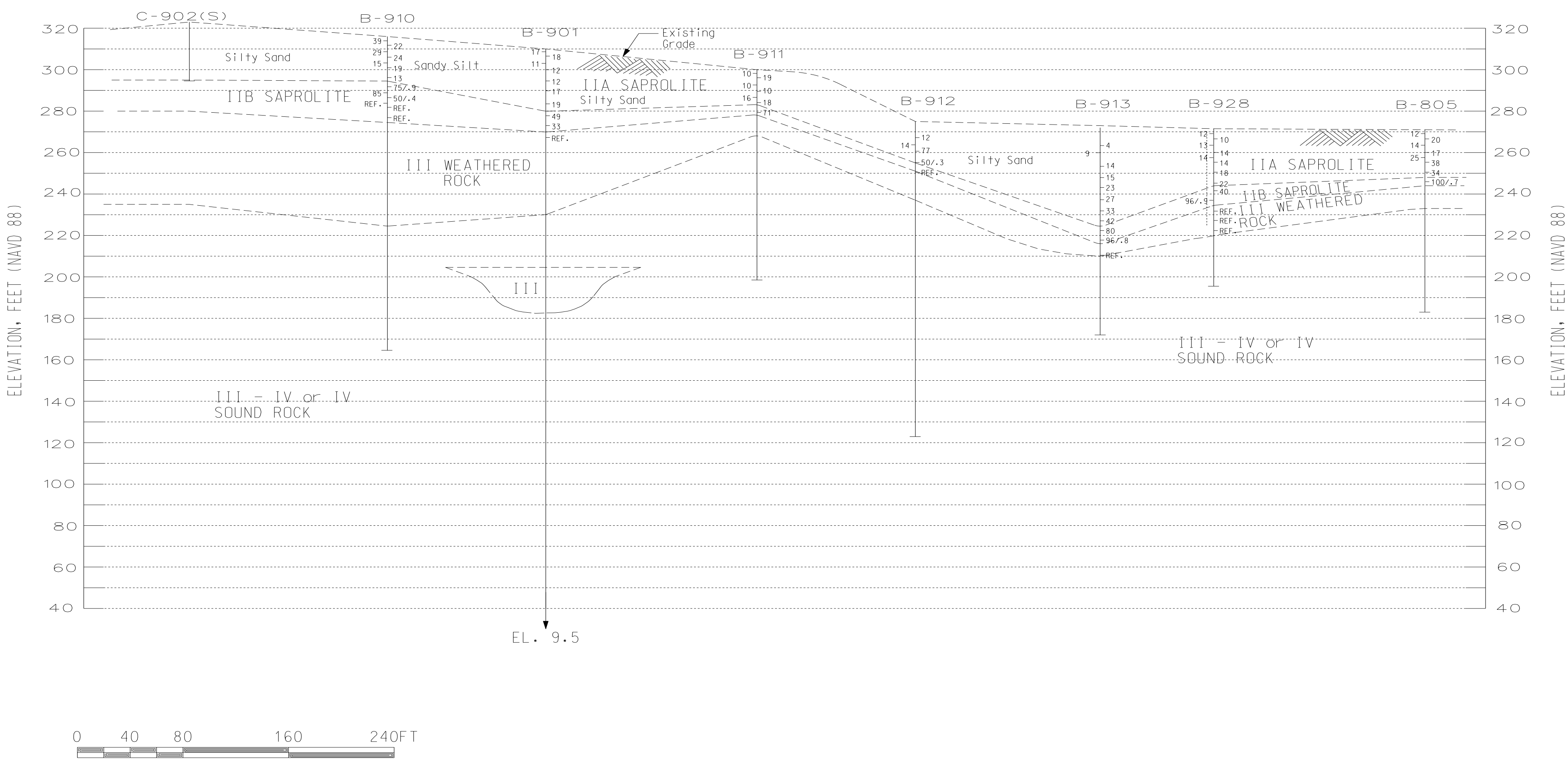
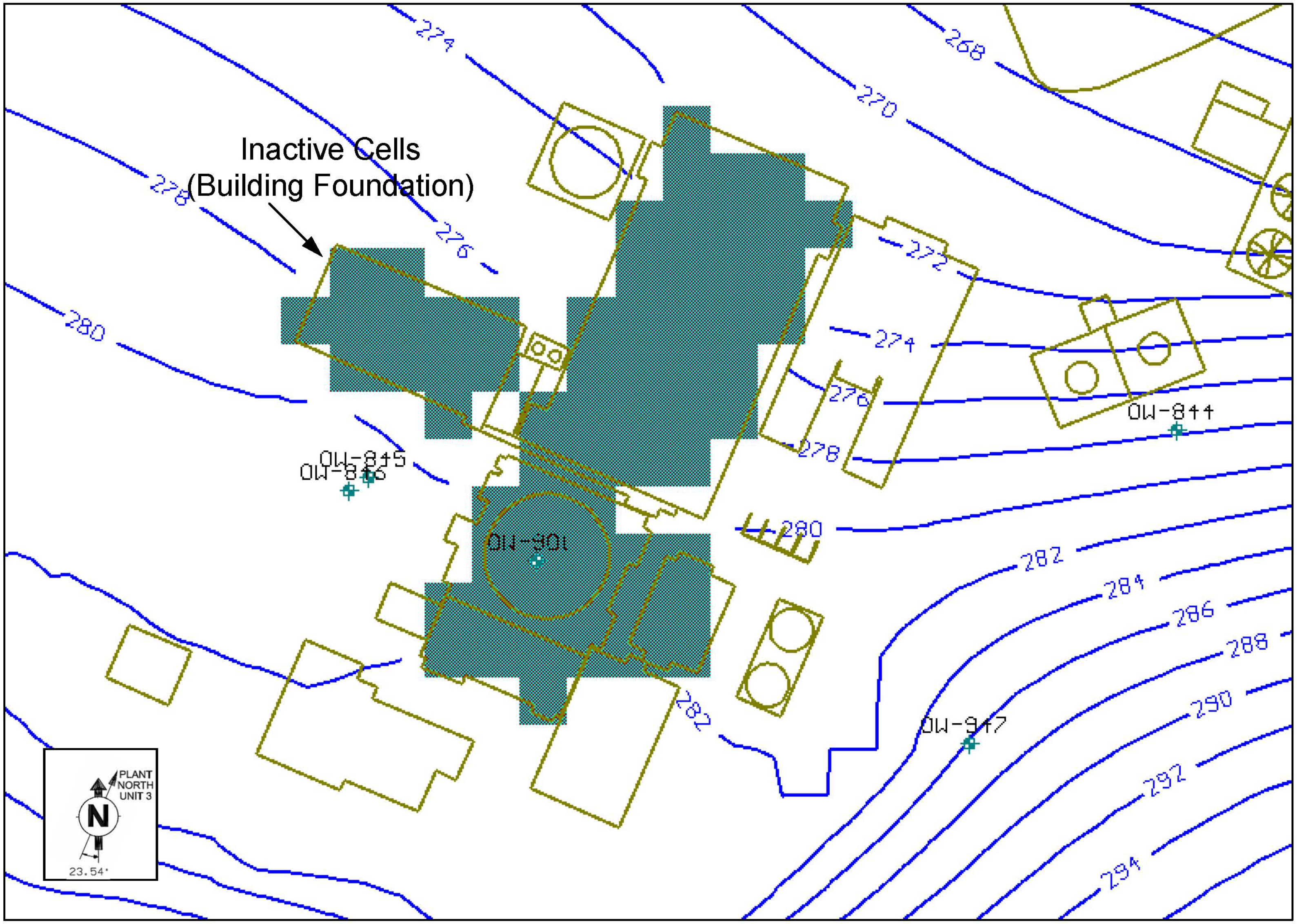
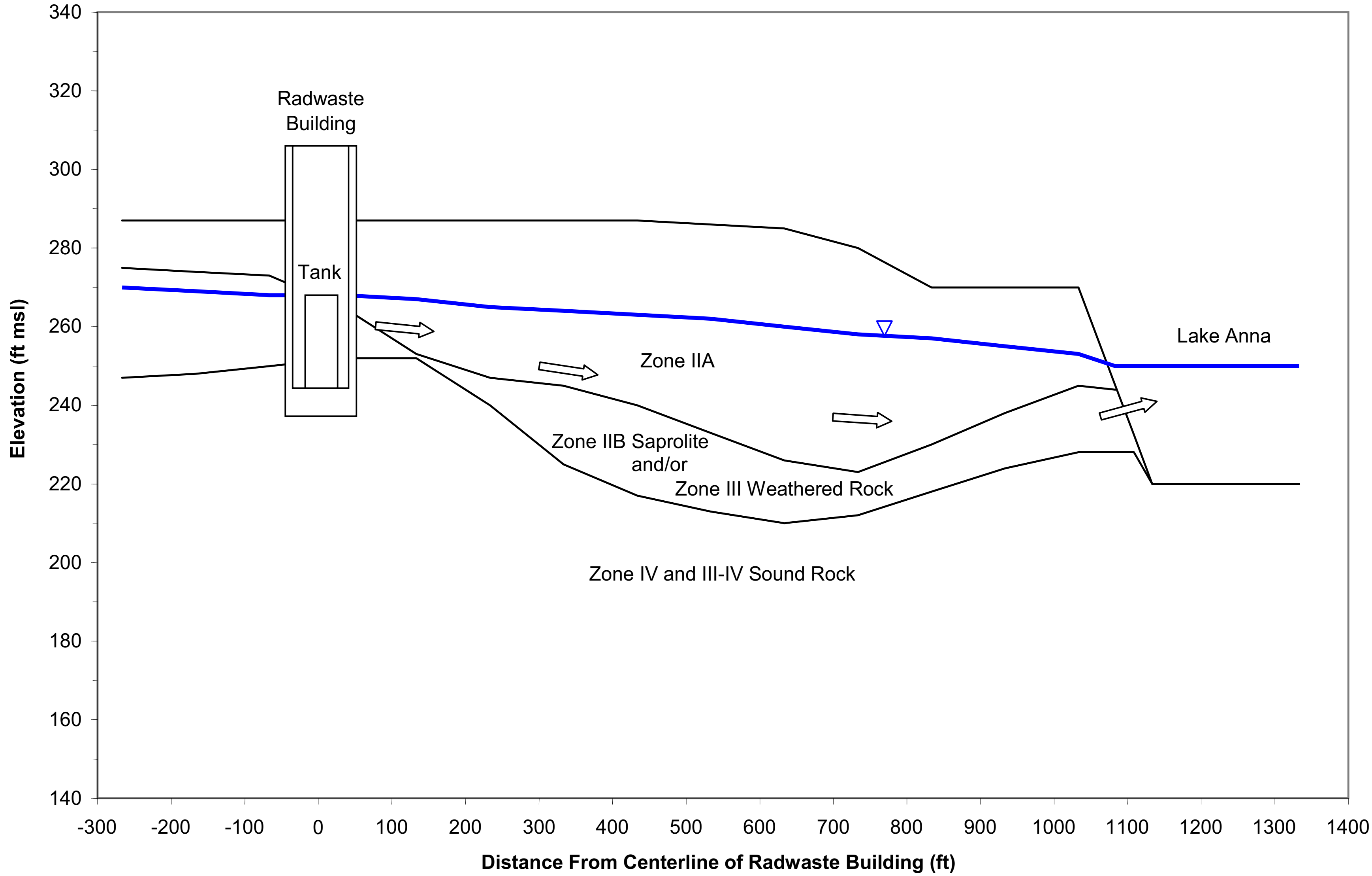


Figure 2.4-216 Piezometric Head Contour Map of Post-Construction Groundwater Elevation Contours Around the Unit 3 Power Block (contours in ft)



NAPS COL 2.0-24-A **Figure 2.4-217 Model for Evaluating Radionuclide Transport in Groundwater**



2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Basic Geologic and Seismic Information

NAPS COL 2.0-26-A

The information needed to address DCD COL Item 2.0-26-A is included in SSAR Section 2.5.1, which is incorporated by reference with the following supplements.

2.5.1.2.3 Site Area Stratigraphy

The third paragraph of this SSAR section is supplemented as follows with information that addresses the geological and geotechnical data collected from the additional Unit 3 borings.

NAPS COL 2.0-26-A

Seven borings were completed to depths ranging between 15 and 52 m (50 and 170 ft) during the ESP investigation (SSAR Appendix 2.5.4B). To supplement the existing geological and geotechnical data, 55 borings, 23 cone penetrometer tests (CPTs), 6 test pits, 3 sets of borehole geophysical logging, 3 sets of shear wave suspension logging, and 2 sets of electrical resistivity tests were performed as part of the subsurface investigation program for Unit 3. The boring data and geotechnical testing are discussed in detail in Section 2.5.4. The data developed by the Unit 3 subsurface investigation program are presented in Appendix 2.5.4AA.

b. **Ta River Metamorphic Suite (Cambrian and/or Ordovician)**

The fourth paragraph of Item b of this SSAR section is supplemented as follows with information that summarizes the Unit 3 subsurface investigation program.

Borings completed during previous subsurface investigations at the NAPS site (SSAR References 7 and 8; and SSAR Appendix 2.5.4B) and borings completed as part of the Unit 3 subsurface investigation encountered rocks of the Ta River Metamorphic Suite at the Unit 3 site.

The tenth paragraph of Item b of this SSAR section is supplemented as follows with information describing the results of the subsurface investigation performed for Unit 3.

NAPS ESP COL 2.5-1

Borings completed at the Unit 3 site as part of the Unit 3 subsurface investigation, documented in Appendix 2.5.4AA, encountered the top of the moderately to highly weathered rock (Zone III) from about

Elevation 62.78 to 86.86 m (206 to 285 ft). The maximum thickness of the Zone III rock measured about 23.47 m (77 ft) and is described in the boring logs as a yellowish brown, gray, tan, reddish brown and dark green, very severely to moderately weathered, very closely to closely fractured, very soft to hard, biotite quartz gneiss and quartz biotite gneiss, with traces of clay, iron oxide staining, magnetite, muscovite and feldspar. The top of the slightly weathered to moderately weathered rock (Zone III-IV) was encountered in the borings at elevations ranging from about 56.99 to 89.0 m (187 to 292 feet) and is generally described in the boring logs as a reddish brown to gray, moderately to slightly weathered, very close to moderately fractured, soft to very hard, biotite quartz gneiss and quartz biotite gneiss. The top of the slightly weathered to fresh rock (Zone IV) was encountered in the borings at elevations ranging between about 53.03 to 84.73 m (174 and 278 feet) and is generally described in the boring logs as a gray and reddish brown, slightly weathered to fresh, very close to widely fractured, very hard, biotite quartz gneiss and quartz biotite gneiss.

The last paragraph of Item b of this SSAR section is supplemented with a new paragraph on Unit 3-specific geologic boring results.

The borings revealed highly to moderately weathered rock (Zone III) intervals in the Zone III-IV and Zone IV rock. These intervals were encountered in several of the borings at varying elevations ranging from 87.47 to 47.55 m (287 to 156 ft). The intervals ranged in thickness from about 1.5 to 6.1 m (5 to 20 ft). (Appendix 2.5.4AA)

h. **Residual Soil and Saprolite (Cenozoic)**

Residual Soil

The second paragraph of Item h of this SSAR section is supplemented as follows with information to address residual soil characterization.

Residual soil was not encountered in any of the borings drilled as part of the Unit 3 subsurface investigation. (Appendix 2.5.4AA)

Saprolite

The last paragraph of Item h of this SSAR section is supplemented as follows with a new paragraph that addresses geologic findings relative to saprolite.

Borings drilled as part of the subsurface investigation for Unit 3 encountered the top of the Zone IIA saprolite at elevations ranging from about 70.71 to 102.11 m (232 to 335 ft). The thickest Zone IIA saprolite encountered was about 17.98 m (59 ft) while the median thickness was about 7.62 m (25 ft). The saprolite is generally described in the boring logs as a yellowish red and reddish yellow clayey silt, silty sand and sand with relict rock fabric. The top of the Zone IIB saprolite was encountered at elevations ranging from about 65.53 to 91.74 m (215 to 301 ft). The thickest Zone IIB saprolite encountered was about 11.88 m (39 ft) while the median thickness was about 2.74 m (9 ft). The saprolite is generally described in the boring logs as a pale yellow to gray to orange brown, silty, fine to coarse sand and very severely weathered, soft to moderately hard gneiss with traces of clay, mafic minerals, and iron oxide staining.

k. **Artificial Material**

The first paragraph of Item k of this SSAR section is supplemented as follows with information to address findings relative to artificial material.

Borings performed as part of the subsurface investigation for Unit 3 encountered fill to depths of between about 0.12 to 5.48 m (0.4 and 18 ft) below the ground surface. The maximum thickness of fill (18 ft) was encountered in boring B-932 and is described in the boring log as a greenish gray and yellowish brown sandy silt and clay with traces of gravel and organic debris. (Appendix 2.5.4AA)

The first paragraph of Item k of this SSAR section is supplemented with information on prohibiting the use of Zone IIA soil as structural fill.

NAPS ESP PC 3.E(5)
NAPS ESP VAR 2.5-2

As described in Section 2.5.4.5.3, Zone IIA soil will not be used as structural fill to support Seismic Category I or II structures.

2.5.1.2.6 Site Engineering Geology Evaluation

a. **Engineering Behavior of Soil and Rock**

Soil

The second paragraph under Soil in Item a of this SSAR section is supplemented as follows with information to address soil behavior.

NAPS COL 2.0-26-A

The saprolite at the Unit 3 site has been categorized into Zone IIA and Zone IIB saprolite, based on its general composition and grain size (Section 2.5.4). Grain size tests on samples of the Zone IIA saprolite show that the median fines content for the saprolite is about 25 percent with the majority of the samples classified as a silty sand (SM). Grain size tests on samples of the Zone IIB saprolite show that the fines content for the saprolite ranges from about 15 to 25 percent. The saprolite is also classified as a silty sand (SM). Zone IIA saprolite is the more weathered of the two saprolites and contains less than 10 percent rock fragments with relict texture. The borings drilled as part of the subsurface investigation for Unit 3, documented in Appendix 2.5.4AA, reveal that SPT N-values ranged from 2 to refusal, with a median value of 15 blows per foot (bpf) for this saprolite. Zone IIB saprolite contains between 10 and 50 percent relict rock fragments, and SPT N-values ranged from 24 to refusal with a median value of 75 bpf. Section 2.5.4 contains a detailed discussion of the geotechnical properties of the saprolite at the Unit 3 site.

Rock

The second paragraph under Rock of Item a of this SSAR section is supplemented as follows with information to address rock behavior.

Based on the results of the borings drilled as part of the subsurface investigation for Unit 3, documented in Appendix 2.5.4AA, rock quality designation (RQD) generally ranges from zero to 50 percent for the Zone III rock with an average RQD value of about 20 percent. An RQD of 20 percent is indicative of very poor quality rock (SSAR Reference 109).

The third paragraph under Rock of Item a of this SSAR section is supplemented as follows with information to address rock behavior.

Based on the results of the borings drilled as part of the subsurface investigation for Unit 3 and documented in Appendix 2.5.4AA, RQD generally ranges from about 50 to 90 percent for the Zone III-IV rock with an average value of about 65 percent, indicative of fair quality rock (SSAR Reference 109). For the Zone IV rock, RQD is generally above 80 percent and mostly above 90 percent. The average RQD value is 95 percent, indicative of excellent quality rock (SSAR Reference 109). The boring results for the previous geotechnical investigations (SSAR References 7 and 8), and for both the ESP subsurface investigation (Reference 2.5-201) and the Unit 3 subsurface investigation (Appendix 2.5.4AA) indicate that Zones III-IV and IV are suitable bearing surfaces on which to found the Seismic Category I structures. The joints and fractures present in both zones are not of sufficient density or areal extent to affect the engineering behavior of the rock with respect to its foundation bearing capacity or integrity.

b. **Zones of Alteration, Weathering and Structural Weakness**

The fourth paragraph of Item b of this SSAR section is supplemented as follows with information on excavation and replacement of weathered or fractured rock.

NAPS ESP PC 3.E(4)

Weathered or fractured rock at the foundation level for safety-related structures will be excavated and replaced with lean concrete before initiation of foundation construction. See also Section 2.5.4.10.

f. **Construction Groundwater Control**

The first paragraph of Item f of this SSAR section is supplemented as follows with information to address ground water level.

Groundwater levels at the site are expected to result in the need for temporary dewatering of foundation excavations extending below the water table. Dewatering will be performed in a manner that minimizes drawdown effects on the surrounding environment. Drawdown effects will be limited to the Unit 3 site and no offsite users will be affected.

g. **Unforeseen Geologic Features**

The first paragraph of Item g of this SSAR section is supplemented as follows with information to address geologic mapping of excavations of safety-related structures.

NAPS ESP PC 3.E(6)

Future excavations for safety-related structures will be geologically mapped. Unforeseen geologic features that are encountered will be evaluated. The NRC will be notified no later than 30 days before any excavations for safety-related structures are open for NRC examination and evaluation. See also Section 2.5.4.5.2.

2.5.1.2.7 Site Groundwater Conditions

The second paragraph of this SSAR section is supplemented as follows with information to address site groundwater conditions.

NAPS COL 2.0-26-A A detailed discussion of Unit 3 site groundwater conditions based on the Unit 3 subsurface investigation is provided in Section 2.4.12.

2.5.2 Vibratory Ground Motion

NAPS COL 2.0-27-A The information needed to address DCD COL Item 2.0-27-A is included in SSAR Section 2.5.2, which is incorporated by reference with the following variances and supplements.

2.5.2.5 Seismic Wave Transmission Characteristics of the Site

The third paragraph in this SSAR section is supplemented as follows with information to address the materials under the foundations of the Seismic Category I structures for Unit 3.

NAPS COL 2.0-27-A The Reactor Building/Fuel Building (RB/FB) and the Control Building (CB) are founded on sound bedrock, both Zone IV and Zone III-IV. The FWSC is founded on Zone III weathered rock and structural fill.

The fourth paragraph in this SSAR section is supplemented as follows with information to address the seismic wave transmission characteristics of site materials under Unit 3.

The seismic wave transmission characteristics of the site materials are described in Section 2.5.4.7. The description includes the shear wave velocity profile for the Unit 3 site and the variation of shear modulus and damping with strain for Zone II and III materials above the sound bedrock. Shear wave velocity profiles for rock and soil under Unit 3 are described in Section 2.5.4.7. The shear wave velocity profiles extend from design plant grade at an elevation of 88.4 m (290 ft) to over 30 m (100 ft) below the depth at which the bedrock under the site reaches a velocity of about 2.80 km/s (9200 fps). The shear wave profile of bedrock is used to evaluate amplification of the 2.80 km/s (9200 fps) hard rock SSE ground motion to the top of competent rock, selected to be at the top of the Zone III-IV material (Elevation 83.2 m (273 ft)), with a best-estimate shear wave velocity of 1.28 km/s (4200 fps). Note that this best estimate is less than the best estimate value given in Table 2.5-212, for Zone III-IV rock, because there is some Zone III weathered rock present at Elevation 83.2 m (273 ft). Also, because the subsurface investigation for Unit 3 was performed specific to the locations of the RB/FB, CB, and FWSC, the data obtained on site materials resulted in a change in the control point elevation from 76.2 m (250 ft) to 83.2 m (273 ft). The change in control point, along with the change in control point SSE response spectra, is a variance from the SSAR. Free-field outcrop ground motions at two additional horizons within this profile are also evaluated; one at the base of the foundation for the CB and the other at the base of the foundation for the RB/FB (at elevations of 73.5 m (241 ft) and 68.3 m (224 ft), respectively).

NAPS ESP VAR 2.0-4

The fourth paragraph in this SSAR section is further supplemented to address the subsurface profile of seismic wave transmission characteristics for the FWSC as follows.

The subsurface profile of the above analyses was supplemented to include material between the top of competent material under the FWSC (Elevation 72.2 m (237 ft)) and the base of the foundation (Elevation 86.0 m (282 ft)) for analysis of ground motions for the dynamic design of the FWSC.

The fifth paragraph of this SSAR section is supplemented as follows with information to address the subsurface profile of seismic wave transmission characteristics for Unit 3 areas outside of the power block.

Finally, a thicker soil profile of in situ material above the 83.2 m (273 ft) elevation is used to evaluate liquefaction potential and slope stability at the site. Section 2.5.4.7.3 and Section 2.5.4.7.4 describe the site-specific acceleration-time histories developed for the hard rock SSE and the results of rock and soil column amplification/attenuation analyses.

2.5.2.6.7 Selected SSE Ground Motion

c. Selection of Enveloping Horizontal SSE Spectrum

The sixth paragraph of Item c in this SSAR section is supplemented as follows with information to address the subsurface shear wave velocity for the Unit 3 site.

NAPS COL 2.0-27-A

Section 2.5.4.7 describes site-specific subsurface shear wave velocity and related material property information for the site. Based on these data, a site shear wave velocity profile has been developed. This profile has been used to calculate the amplification by subsurface material above the 2.80 km/s (9200 fps) hard rock Unit 3 site SSE ground motion at a control point located on the top of competent Zone III-IV rock. As noted in Section 2.5.2.5, a shear wave velocity for the Zone III-IV material of 1.28 km/s (4200 fps) has been used in the control point SSE analysis. The elevation of the top surface of the Zone III-IV material varies across the site, as shown in the six subsurface profiles in Figure 2.5-215 through Figure 2.5-220. The top of the Zone III-IV material has been chosen to be at an elevation of 83.2 m (273 ft) in the control point SSE analysis.

The seventh paragraph of Item c in this SSAR section is supplemented as follows with information to address the subsurface shear wave velocity for the Unit 3 site.

Both high frequency and low frequency time histories were developed for the evaluation of the effect of site-specific subsurface shear wave velocities between the 2.80 km/s (9200 fps) and 1.28 km/s (4200 fps) control points. These time histories were made to match spectra that, in composite, matched the hard rock SSE spectrum but that, individually, are based on the high and low frequency reference probability response spectra shapes.

The ninth paragraph of Item c in this SSAR section is supplemented as follows with information to address the DBE stochastic model for the Unit 3 site.

A stochastic model described in SSAR Reference 170, with some modifications to account for the conditions at the Unit 3 site, was used to generate 60 randomizations of the Unit 3 site-specific rock column velocity profile between elevations with shear wave velocities of 2.80 km/s (9200 fps) and 1.28 km/s (4200 fps).

The tenth paragraph of Item c in this SSAR section is supplemented as follows with information to describe the inputs to the SHAKE2000 computer runs for the Unit 3 site.

A set of SHAKE2000 runs was performed on each of the 60 randomized rock profiles using the two input hard rock motions. The site was modeled by horizontal layers overlying a uniform half-space of hard bedrock subjected to the vertically propagating shear wave time histories. The response spectra from the SHAKE2000 analyses were defined at 301 frequencies from 0.1 to 100 Hz. The enveloped log-average spectrum for the Zone III-IV hypothetical rock outcrop control point at Elevation 83.2 m (273 ft) and shear wave velocity of 1.28 km/s (4200 fps) was fit with a smooth fitting function. See Figure 2.5-201. The resultant fitting function was used to obtain the response spectrum for the same set of 21 frequencies as used in the SSAR. This 21-frequency set of response spectral ordinates defines the rock response spectrum for the corresponding hypothetical rock outcrop control point on the top of Zone III-IV material. This horizontal spectrum is shown in Figure 2.5-205.

The last paragraph of Item c of this SSAR section is supplemented as follows with two new paragraphs to address the output to the SHAKE2000 computer runs for the Unit 3 site.

Output from the same SHAKE2000 runs was also collected and used to develop smooth horizontal free-field outcrop motions at elevations corresponding to the bases of the foundations of the CB and RB/FB (73.5 m (241 ft) and 68.3 m (224 ft), respectively). The SHAKE2000 results and derived smooth fitting functions for these elevations are shown in Figure 2.5-202 and Figure 2.5-203. These horizontal spectra are shown in Figure 2.5-206 and Figure 2.5-207.

Finally, SHAKE2000 runs were performed incorporating the material properties up to the base of the foundation of the FWSC. Again, smooth free-field horizontal spectra were developed in the same way for this elevation. See Figure 2.5-204 and Figure 2.5-208.

d. **Development of Vertical SSE Spectra**

Zone III-IV Hypothetical Rock Outcrop Control Point SSE Spectrum

The third paragraph of Item d of this SSAR section is supplemented as follows to address the horizontal response spectrum and elevation at the top of competent material for Unit 3 site.

The horizontal SSE spectral accelerations, V/H ratios, and vertical SSE spectral accelerations for the Zone III-IV hypothetical rock outcrop control point are listed in Table 2.5-201. The vertical SSE spectrum is calculated by multiplying the selected horizontal SSE spectral amplitude at each frequency by the applicable V/H ratio for that frequency from NUREG/CR-6728 (SSAR Reference 171). The selected horizontal and vertical spectra at the top of competent material at Elevation 83.2 m (273 ft) are plotted in Figure 2.5-205.

The third paragraph of Item d of this SSAR section is supplemented as follows with two new paragraphs to address the foundation horizon for Unit 3 Seismic Category I structures.

CB and RB/FB Foundation Horizon Spectra

The horizontal SSE spectral accelerations, V/H ratios, and vertical SSE spectral accelerations for the CB and RB/FB foundation horizons are listed in Table 2.5-202 and Table 2.5-203, respectively. The vertical SSE spectrum is calculated by multiplying the selected horizontal SSE spectral amplitude at each frequency by the applicable V/H ratio for that frequency from SSAR Reference 171. The selected horizontal and vertical spectra at the base of the CB and RB/FB foundation elevations are plotted in Figure 2.5-206 and Figure 2.5-207, respectively.

FWSC Foundation Spectra

The horizontal SSE spectral accelerations, V/H ratios, and vertical SSE spectral accelerations for the ground surface at the FWSC location are listed in Table 2.5-204. The vertical SSE spectrum is calculated by multiplying the selected horizontal SSE spectral amplitude at each frequency by the applicable V/H ratio for that frequency from SSAR Reference 171. The selected horizontal and vertical spectra for the ground surface at the location of the FWSC are plotted in Figure 2.5-208.

2.5.2.6.8 Additional Sensitivity Studies

The last paragraph of this SSAR section is supplemented with a new paragraph on sensitivity studies.

NAPS COL 2.0-27-A

The SSAR sensitivity analyses for the reference probability and performance-based approaches were not re-performed for the FSAR.

2.5.2.6.9 Additional Modification of the Selected Spectrum

The last paragraph of this SSAR section is supplemented as follows with information explaining why additional modification of the selected spectrum is unnecessary for Unit 3.

NAPS COL 2.0-27-A

The potential modifications to the selected spectrum were not performed for Unit 3 because, as shown in Table 2.0-201, the certified seismic design response spectra (CSDRS) for Seismic Category I structures bound the high-frequency content in the foundation input response spectra (FIRS).

2.5.2.6.10 Approach to Develop the EDS

The last paragraph of this SSAR section is supplemented as follows with information explaining why additional modification of the selected spectrum is unnecessary for Unit 3.

NAPS COL 2.0-27-A

The potential modifications to the selected spectrum described in SSAR Section 2.5.2.6.9 were not performed for Unit 3 because, as shown in Table 2.0-201, the CSDRS for Seismic Category I structures bound the high-frequency content in the FIRS.

2.5.2.7 Operating Basis Earthquake

This SSAR section is supplemented as follows with information regarding the operating basis earthquake.

NAPS COL 2.0-27-A

The comparison of CSDRS and FIRS for Seismic Category I structures is provided in Section 2.0. The DCD OBE ground motion is chosen to be one-third of the CSDRS per DCD Section 3.7.1. Consistent with SSAR Section 2.5.2.7, the Unit 3 OBE ground motion would be one-third of the FIRS. Because one-third of the CSDRS exceeds one-third of the FIRS, the DCD OBE bounds the site OBE.

2.5.3 Surface Faulting

NAPS COL 2.0-28-A

The information needed to address DCD COL Item 2.0-28-A is included in SSAR Section 2.5.3, which is incorporated by reference with the following supplements.

NAPS COL 2.0-28-A

2.5.3.2.5 Unit 3 Subsurface Investigation

Borehole data, from the supplemental subsurface investigation described in Section 2.5.4.3, were reviewed for evidence of Quaternary fault movement. No such evidence was exhibited by the borehole data.

2.5.4 Stability of Subsurface Materials and Foundations

NAPS COL 2.0-29-A

The information needed to address DCD COL Item 2.0-29-A is included in SSAR Section 2.5.4, which is incorporated by reference with the following supplements.

SSAR Section 2.5.4 has been supplemented by integrating information on the additional Unit 3 borings into a single section with the same numbering as the SSAR.

2.5.4.1 Geologic Features

SSAR Section 2.5.1.1 describes the regional geology, including regional physiography and geomorphology, regional geologic history, regional stratigraphy, and the regional tectonic setting. SSAR Section 2.5.1.2 addresses site-specific geology and structural geology, including site physiography and geomorphology, site geologic history, site stratigraphy, site structural geology, and a site geologic hazard evaluation.

2.5.4.2 Properties of Subsurface Materials

2.5.4.2.1 Introduction

This section describes the static and dynamic engineering properties of the Unit 3 site subsurface materials. An overview of the subsurface profile and materials is given in Section 2.5.4.2.2. The field investigations are described in Section 2.5.4.2.3. The laboratory tests on soil and rock samples from the investigation and their results are presented in Section 2.5.4.2.4. The engineering properties of the subsurface materials are given in Section 2.5.4.2.5.

2.5.4.2.2 Description of Subsurface Materials

The following is a brief description of the subsurface materials, giving the soil and rock constituents, and their range of thicknesses encountered at the Unit 3 site. The information was taken from the 55 borings made at the site (outlined in Section 2.5.4.2.3). For reference, the existing site ground surface elevations in the areas explored range from about Elevation 76.2 m (250 ft) to Elevation 102.1 m (335 ft), with a median of about Elevation 90.2 m (296 ft). The design grade elevation for Unit 3 is Elevation 88.4 m (290 ft).

a. **Zone IV Bedrock**

The Unit 3 subsurface investigation (Appendix 2.5.4AA) describes the bedrock underlying the power block area mostly as quartz gneiss, biotite quartz gneiss, quartz biotite gneiss, or biotite gneiss. A detailed description of the bedrock is provided in Section 2.5.1.2.3.

The top of Zone IV bedrock encountered in the borings made for Unit 3 ranges from about Elevation 53.0 m (174 ft) to Elevation 84.7 m (278 ft). Top of Zone IV rock contours beneath the Unit 3 power block area are shown on Figure 2.5-209. The top of Zone III-IV bedrock ranges from about Elevation 57.0 m (187 ft) to Elevation 89.0 m (292 ft). Top of Zone III-IV rock contours beneath the Unit 3 power block area are shown on Figure 2.5-210.

b. **Zone III Weathered Rock**

The top of Zone III bedrock encountered in the borings made for Unit 3 ranges from about Elevation 62.8 m (206 ft) to Elevation 86.9 m (285 ft). The maximum thickness measured is about 23.5 m (77 ft). Top of Zone III rock contours beneath the Unit 3 power block area are shown on Figure 2.5-211.

c. **Zone IIA and IIB Saprolites**

Distribution of Zone IIA and IIB saprolites varies throughout the Unit 3 site. The Zone IIB saprolites represent about 30 percent of the saprolites on site and are typically very dense silty sands with from 10 to 50 percent core stone. The thickest Zone IIB deposit encountered in the Unit 3 borings was 11.9 m (39 ft) while the median thickness was about 2.7 m (9 ft). The top of Zone IIB saprolite encountered ranges from about Elevation 65.5 m (215 ft) to Elevation 91.7 m (301 ft). Top of Zone IIB saprolite contours beneath the Unit 3 power block area are shown on Figure 2.5-212.

The overlying Zone IIA saprolites comprise, at the Unit 3 site, about 70 percent of the saprolitic materials on site. About 80 percent of the Zone IIA saprolites are classified as coarse grained (sands, silty sands), while the remainder are fine grained (clayey sands, sandy and clayey silts, and clays). The thickest Zone IIA deposit encountered in the Unit 3 borings was 18.0 m (59 ft) while the median thickness was about 7.6 m (25 ft). The top of Zone IIA saprolite ranges from about Elevation 70.7 m (232 ft) to Elevation 102.1 m (335 ft). Top of Zone IIA saprolite contours beneath the Unit 3 power block area are shown on Figure 2.5-213.

d. **Zone I and Fill**

For Unit 3 foundations, Zone I soils and existing fills will be excavated. Thus, they are not considered further here.

e. **Subsurface Profiles**

Figure 2.5-215 through Figure 2.5-220 illustrate typical subsurface profiles across the Unit 3 power block area. The locations of these profiles are shown in Figure 2.5-214. These profiles, with structure cross-sections added, are presented to illustrate foundation interfaces in Section 2.5.4.3. They also are used to illustrate the Unit 3 excavation in Section 2.5.4.5, and for bearing capacity considerations in Section 2.5.4.10.

2.5.4.2.3 Field Investigations

The borings, observation wells, and cone penetrometer tests from the Unit 3 site exploration program are summarized in Table 2.5-205, Table 2.5-206, and Table 2.5-207, respectively. The elevations, depths and thicknesses of the subsurface zones observed from the individual borings are shown in Table 2.5-208. Geophysical surveys are described in Section 2.5.4.4.

The subsurface field investigation was performed during August through November 2006. The majority of the investigation was conducted in the power block area with the number and depth of investigation points conforming to the guidance provided in RG 1.132 (SSAR Reference 153). Additional exploration points were located outside the power block area, e.g., at the proposed locations for the cooling towers.

The Unit 3 exploration point locations in the power block area are shown in Figure 2.5-221. Borings from previous exploration programs are also shown. Exploration points outside the power block area are shown on Figure 2.5-222.

The scope of work and the special methods used to collect field data are listed below:

- 55 exploratory borings (MACTEC Engineering and Consulting, Raleigh, North Carolina)
- 7 observation wells with permeability (slug) tests in 4 wells (MACTEC Engineering and Consulting, Raleigh, North Carolina, and Bedford Well Drilling, Bedford, Virginia)
- 4 packer tests (Miller Well Drilling, Hayesville, North Carolina, under MACTEC supervision)
- 23 CPTs plus 4 down-hole seismic cone tests and pore pressure dissipation tests in 4 CPTs (Gregg InSitu, Inc., Columbia, South Carolina)
- 6 test pits (MACTEC Engineering and Consulting, Raleigh, North Carolina)
- 3 sets of borehole geophysical logging and 3 sets of suspension P-S velocity logging (GEOVision, Corona, California)
- 2 sets of electrical resistivity tests (MACTEC Engineering and Consulting, Raleigh, North Carolina)
- Survey of exploration points (McKim and Creed, Virginia Beach, Virginia)

The exploration program was performed using the guidance in RG 1.132 (SSAR Reference 153). The fieldwork was performed under an audited and approved quality assurance program and work procedures developed specifically for the Unit 3 project. MACTEC Engineering and Consulting, contracted to Dominion to perform the subsurface investigation, worked under MACTEC's Quality Assurance Plan that met the requirements of 10 CFR 50, Appendix B. This Plan included meeting the requirements of Subpart 2.20 of ASME NQA-1, 1994 edition (Reference 2.5-204).

The subsurface investigation and sample/core collection was directed by the MACTEC site manager who was on site at all times during the field operations. A Bechtel geotechnical engineer or geologist, along with a Dominion representative, was also on site continuously during these operations. MACTEC's QA/QC engineer was on site part of the time. The draft boring and well logs were prepared in the field by MACTEC geologists.

Sample and core storage and handling were in accordance with ASTM D 4220 (Reference 2.5-205). An on-site storage facility for soil samples and rock cores was established before the fieldwork began. This facility was in the limited access and climate controlled “A” Level area of the Units 1 and 2 warehouse. Samples and cores were stored either within a 3.7 m (12 ft) square area surrounded by a 1.8 m (6 ft) high chain link fence, or in an adjacent secured area. Each sample and core was logged into an inventory control system. Samples removed from the facility were noted in the sample inventory logbook. A chain-of-custody form was also completed for samples removed from the facility.

Details and results of the exploration program are contained in Appendix 2.5.4AA. The borings, observation wells, CPTs and test pits are summarized below. The laboratory tests are summarized and the results presented in Section 2.5.4.2.4. The geophysical tests are summarized and the results presented in Section 2.5.4.4.

a. **Borings and Samples/Cores**

The 55 borings drilled ranged from 6.7 m (22 ft) to 91.4 m (300 ft) in depth. The 91.4 m (300 ft) deep boring was drilled at the center of the Reactor Building (RB) location, to about 65.5 m (215 ft) depth in sound rock beneath the bottom of the basemat level. The borings were advanced in soil using rotary wash drilling techniques until standard penetration test (SPT) refusal (defined as 50 blows per 25 mm (1 in) or less for start of rock coring) occurred. Steel casing was then set into the rock, and the holes were advanced using wireline rock coring equipment consisting of a 1.5 m (5 ft) long “HQ” core barrel with a split inner barrel.

The soil was sampled using an SPT sampler at 0.76 m (2.5 ft) intervals to about 4.6 m (15 ft) depth and at 1.5 m (5 ft) intervals below 4.6 m (15 ft). The SPT was performed using an automatic hammer, and was conducted in accordance with ASTM D 1586 (SSAR Reference 155). The recovered soil samples were visually described and classified by the onsite geologist. A selected portion of the soil sample was placed in a glass sample jar with a moisture-proof lid. The sample jars were labeled, placed in boxes, and transported to the on-site storage area.

Energy measurements were made on the automatic SPT hammers used by the four drill rigs that performed the borings. The energy measurements were made in accordance with ASTM D 4633 (Reference 2.5-206). The average energy transfer ratio (ETR) for each rig ranged from 75.2 percent to 82.8 percent, with an overall average of 79.2 percent. The N-values shown on the boring logs (Appendix 2.5.4AA) and on the subsurface profiles (Figure 2.5-215 through Figure 2.5-220) are not adjusted for hammer energy. N-values used in engineering analysis (e.g., liquefaction analysis) are adjusted for hammer energy, i.e., N_{60} was used in these situations.

Undisturbed samples were obtained in accordance with ASTM D 1587 (Reference 2.5-220) using a Shelby tube sampler or a rotary Pitcher sampler. Upon sample retrieval, the disturbed portions at both ends of the tube were removed, both ends were trimmed square to establish an effective seal, and pocket penetrometer (PP) tests were performed on the trimmed lower end of the samples. Both ends of the sample were then sealed with hot wax, covered with plastic caps, and sealed once again using electrician tape and wax. The tubes were labeled and transported to the sample storage area. Undisturbed samples are identified on the boring logs included in Appendix 2.5.4AA.

Rock coring was performed in accordance with ASTM D 2113 (SSAR Reference 156). After removal from the split inner barrel, the recovered rock was carefully placed in wooden core boxes. The onsite geologist visually described the core, noting the presence of joints and fractures, and distinguishing natural breaks from mechanical breaks. The geologist also computed the percentage recovery and the RQD. Photographs of the cores were taken in the field. Filled and labeled core boxes were transported to the on-site sample storage facility.

The boring logs and the photographs of the rock cores are provided in Appendix 2.5.4AA, along with details of the automatic hammer energy measurements. Borehole locations, depths, etc. are summarized in Table 2.5-205. The soil and rock materials encountered in the Unit 3 borings were similar to those found in the previous sets of borings conducted at the NAPS site. The elevations, depths and thicknesses of the subsurface zones observed from the individual borings are shown in Table 2.5-208.

b. **Observation Wells**

Each of the seven observation wells was installed adjacent to a sample boring. Three of the wells were screened in the soil/weathered rock zone, while four were screened in rock. Each well depth was selected in the field after a review of the borehole record. For the wells screened in rock, the screen depth was also based on the rock core description and packer test results. Boreholes for the wells in soil/weathered rock were advanced with hollow stem augers while the boreholes for all but one of the wells in rock were advanced using air-rotary drilling techniques. The borehole for the fourth well in rock (OW-951) was advanced with hollow stem augers until auger refusal, and was completed in rock using an "HQ" core barrel with a split inner barrel. This was after repeated cave-ins during attempts to advance the hole with air-rotary drilling.

After the designated depth of each well was reached, and the PVC screen and casing set, the sand pack and bentonite seal were placed, and then a grout plug was placed from the top of the bentonite seal to the ground surface. (In OW-951, a filter sock was placed over the screen, above which a formation packer and bentonite seal were set.) Each well was capped with a lockable steel cap and surrounded with a concrete pad.

Each well was developed by pumping. Two or three standing well volumes of water were purged initially by pumping, cycling the pump on and off to create a surging effect. The well was considered developed when the pH and conductivity stabilized and the pumped water was reasonably free of suspended sediment.

Permeability tests were performed in each of the three wells screened in soil/weathered rock, and in one of the wells screened in rock (OW-949) in accordance with ASTM D 4044, Section 8 (SSAR Reference 157) using a procedure that is commonly termed the slug test method. Slug testing involves establishing a static water level, lowering a solid cylinder (slug) into the well to cause an increase in water level in the well, and monitoring the time rate for the well water to return to the pre-test static level. The slug is then rapidly removed to lower the water level in the well, and the time rate for the water to recover to the pre-test static level is again measured. Electronic transducers and data loggers were used to measure the water levels and times during the test.

Permeability testing by the packer method was conducted in the borings adjacent to the four wells screened in rock. Test procedures used are described in ASTM D 4630 (Reference 2.5-207), as modified by U.S. Army Corps of Engineers in their Rock Testing Handbook (Reference 2.5-208) to use a manually read flowmeter rather than a digitally recorded one. The packer testing method, known as the constant head injection test, involved establishing and maintaining a constant pressure in the test length, measured by an electronic transducer, to determine the rate of inflow associated with maintaining the pressure.

Appendix 2.5.4AA contains the boring logs for the observation wells, the well installation records, the well development records, and the well permeability and packer test results. Observation well locations, depths, etc., are summarized in Table 2.5-206.

c. **Cone Penetrometer Tests**

The 23 CPTs were advanced using a track-mounted 178 kN (20 ton) self-contained cone rig. Each CPT was advanced to refusal, to depths ranging from about 0.91 m (3 ft) to 18.3 m (60 ft). Tip resistance, sleeve friction and porewater pressure were measured. The CPTs were performed in accordance with ASTM D 5778 (SSAR Reference 158). The pore pressure filter was located immediately behind the cone tip.

Down-hole seismic testing was performed at approximately 0.91 m (3 ft) intervals in four of the CPTs (C-902, C-916, C-921 and C-923, see Section 2.5.4.4). One pore pressure dissipation test was performed in each of four CPTS (C-902, C-904b, C-911 and C-917) at depths ranging from about 4.0 m (13 ft) to 8.8 m (29 ft).

The CPT logs, shear wave time of arrival records, and pore pressure versus time plots are contained in Appendix 2.5.4AA. CPT locations, depths, etc., are summarized in Table 2.5-207.

d. **Test Pits**

Six test pits were excavated to depths ranging from about 0.61 m (2 ft) to 1.4 m (4.5 ft) to obtain bulk samples of site soils to test for suitability as backfill. A rubber-tired backhoe was used to excavate the test pits. Bulk samples were collected in new 19 liter (5 gal) plastic buckets. Small portions of the samples were placed in glass jars and sealed for moisture retention.

2.5.4.2.4 Laboratory Testing

Numerous laboratory tests of soil and rock samples were performed for Unit 3. The types and numbers of these tests are shown in Table 2.5-209.

The laboratory testing investigation was performed in accordance with the guidance presented in RG 1.138 (SSAR Reference 148). The laboratory work was performed under an approved quality assurance program with work procedures developed specifically for the Unit 3 project. Soil and rock samples were shipped under chain-of-custody protection from the storage area (described in Section 2.5.4.2.3) to the testing laboratory. When required, samples sent to the testing laboratory were divided and/or shipped to an appropriate testing laboratory under chain-of-custody rules. Laboratory testing of soil and rock samples, except for chemical tests and resonant column torsional shear (RCTS) tests, was performed at the MACTEC laboratories in Charlotte and Raleigh, North Carolina and Atlanta, Georgia. Chemical testing for pH, sulfates and chlorides in selected soil samples was conducted by Severn Trent Laboratories in Earth City, Missouri. RCTS testing of selected soil samples was performed by Fugro Inc. in Houston, Texas, under the technical direction of Dr. K. H. Stokoe of the University of Texas in Austin.

Since the Unit 3 power block area is approximately 460 m (1500 ft) southwest of the center of the Unit 2 Containment Building, the tests focused on verifying that the properties of the soil and rock beneath the Unit 3 power block area were similar to those beneath Units 1 and 2 as determined during previous studies. In addition, chemical tests (for corrosiveness toward buried steel and aggressiveness toward buried concrete) and RCTS tests (for shear modulus and damping ratio variation with cyclic strain) were run on selected saprolite samples.

The details and results of the laboratory testing are included in Appendix 2.5.4AA, except for the RCTS test results which are included in Appendix 2.5.4AAS1. Appendix 2.5.4AA includes references to the industry standards used for each specific laboratory test. The results of the tests on soil samples (excluding strength and RCTS tests) are summarized in Table 2.5-210. Table 2.5-211 gives the results of the unconfined compression tests on the rock cores. The results of the RCTS tests are shown in Figure 2.5-223.

The results of the laboratory tests as they relate to the engineering properties of the soil and rock are described in Section 2.5.4.2.5.

2.5.4.2.5 Engineering Properties

The engineering properties for Zones IIA, IIB, III, III-IV, and IV derived from the Unit 3 field exploration and laboratory testing programs are provided in Table 2.5-212 and described in the following paragraphs. These engineering properties are similar to those obtained from the previous field and laboratory testing programs (as shown in

SSAR Table 2.5-45), with some differences. Where there are differences, the impact from an engineering standpoint is usually either the same or more favorable.

The following paragraphs discuss selected properties shown in Table 2.5-212 under the subheadings: a) rock properties, b) soil properties, c) RCTS results, and d) chemical properties.

a. **Rock Properties**

In general, the rock strength and stiffness values, derived from the field and laboratory testing of the Unit 3 rock, are higher than given in the SSAR. This could reflect less fractured or weathered rock beneath the Unit 3 area, and/or better rock coring equipment and techniques that produced better quality cores.

The Recovery and RQD are based on the results presented for each core in the boring logs in Appendix 2.5.4AA. The RQDs from the borings for Strata III, III-IV and IV are plotted versus elevation in Figure 2.5-224. For Stratum III, RQD generally ranges from zero to around 50 percent, with some higher values. The average value is about 20 percent. For Stratum III-IV, RQD generally ranges from around 50 to 90 percent. The average value is about 65 percent (compared to 50 percent in the SSAR). For Stratum IV, RQD is generally above 80 percent and mostly above 90 percent. The average value is about 95 percent. The average recovery values for Zone III, III-IV and IV are 55 percent, 90 percent, and 98 percent, respectively.

The unconfined compressive strengths and unit weights in Table 2.5-212 are based on the rock strength test results shown in Table 2.5-211. The elastic modulus values are also based on the values shown in Table 2.5-211. The shear modulus values are derived from the elastic modulus values using the Poisson's ratio values tabulated in Table 2.5-212. These higher strain shear modulus values agree well with the low strain values derived from the geophysical tests performed for the Unit 3 exploration program described in Section 2.5.4.4. These high and low strain shear modulus values are essentially the same for high strength rock, certainly for the Zone IV and Zone III-IV rock. Some strain softening has been allowed in the case of the Zone III rock, as described in Section 2.5.4.7. Low strain is defined here as 10^{-4} percent while high strain is taken as 0.25 to 0.5 percent, the amount of strain frequently associated with settlement of structures on soil.

The shear and compression wave velocities in Table 2.5-212 are based on suspension P-S velocity logging performed as part of the Unit 3 exploration program (Appendix 2.5.4AA). These results are summarized in Section 2.5.4.4.4.

b. **Soil Properties**

Zone IIA Saprolite

Grain size curves from sieve analyses of Zone IIA silty and clayey sand, and sandy silt samples are shown in Appendix 2.5.4AA. The tests were run mainly on the silty sand samples with more than 90 percent having fines contents of less than 50 percent. Figure 2.5-225 shows fines content versus depth from these tests. The median fines content for the Zone IIA saprolite is about 25 percent, with the majority of samples having a Unified Soil Classification System (USCS) classification (Reference 2.5-209) of SM.

The median natural moisture content from 93 tests performed is 19 percent. For the relatively small percentage of samples that exhibited plasticity, the median liquid limit was 34 percent while the plasticity index was 11 percent.

The measured SPT N-values from 358 tests ranged from 2 to refusal (defined as >100 blows/0.3 m (1 ft)), with a median value of 15 blows/0.3 m (1 ft). These are plotted versus depth on Figure 2.5-226. The N_{60} median value adjusted for hammer energy is 20 blows/0.3 m (1 ft). The effective angle of internal friction of a medium dense coarse-grained saprolite (N = 20 blows/0.3 m (1 ft)) would typically be taken as around 35 degrees (SSAR Reference 150). However, the relatively high silt content and the presence of low plasticity clay minerals reduce this angle. Consolidated-undrained (C-U) triaxial tests reported in UFSAR Appendices 2C and 3E (SSAR Reference 5) produced internal friction angles (φ') ranging from 23 to 33 degrees, with a median of 30.8 degrees. The average effective cohesive (c') component from the Appendix 2C tests was 13.2 kPa (0.275 kips per square foot (ksf)). A series of C-U tests performed for the Unit 3 program gave effective internal friction angles ranging from about 31 to 36 degrees, with a median of 33 degrees, and very little effective cohesion. The values of φ' = 33 degrees and c' = 6.0 kPa (0.125 ksf) were adopted for the Zone IIA saprolite. This compares with φ' = 30 degrees and c' = 12.0 kPa (0.25 ksf) used in the SSAR.

A large amount of testing was performed after low unit weights were measured in the Zone IIA saprolites in the Units 1 and 2 Service Water Reservoir area. The testing details and results are given in UFSAR Appendix 3E, Attachment 4 (SSAR Reference 5). It was concluded that there are isolated lower densities, but these are not typical. UFSAR Table 3.8-13 (SSAR Reference 5) identifies 125 pcf as a design total unit weight. A value of 19.6 kN/m^3 (125 pcf) is shown in Table 2.5-212.

The shear wave velocities versus depth measured in the soil by suspension P-S velocity logging and CPT seismic testing during the Unit 3 field investigation are shown in Figure 2.5-227. The average shear wave velocity ranges from about 152 m/s (500 feet per second (fps)) to 366 m/s (1200 fps) in the upper 12.2 m (40 ft), with a best estimate of about 259 m/s (850 fps). This is presented in more detail in Section 2.5.4.4 and Section 2.5.4.7.

The high strain (i.e., in the range of 0.25 to 0.5 percent) elastic modulus value has been derived using the relationship with SPT N-value given in SSAR Reference 151. The shear modulus value has been obtained from the elastic modulus values using the relationship between elastic modulus, shear modulus and Poisson's ratio (SSAR Reference 150). The best estimate low strain (i.e., 10^{-4} percent) shear modulus has been derived from the shear wave velocity of 259 m/s (850 fps). The elastic modulus value has been obtained from this shear modulus value using the relationship between elastic modulus, shear modulus, and Poisson's ratio (SSAR Reference 150).

In Table 2.5-212, the value of unit coefficient of subgrade reaction is based on the value for medium dense sand provided by Terzaghi (SSAR Reference 152), while the earth pressure coefficients are Rankine values, assuming level backfill and a zero friction angle between the soil and the wall (see also Section 2.5.4.10.3).

All of the bulk samples obtained from the test pits were Zone IIA saprolite, since the test pits only sampled near-surface soils. Details of the results of the modified Proctor compaction tests and the California Bearing Ratio (CBR) tests run on these samples are provided in Appendix 2.5.4AA. The maximum dry density ranged from about 15.7 kN/m^3 (100 pcf) to 19.8 kN/m^3 (126 pcf), with a median value of 18.2 kN/m^3 (116 pcf). The corresponding optimum moisture content ranged from 9 to 22 percent, with a median value of 13 percent. A plot of molded dry density versus CBR (soaked samples) is given in Figure 2.5-228.

Zone IIB Saprolite

Grain size curves from 15 sieve analyses of Zone IIB silty sand samples are shown in Appendix 2.5.4AA. The samples had fines contents ranging from about 15 to 25 percent. These fines contents are shown versus depth in Figure 2.5-225. The Zone IIB USCS classification is SM.

The measured SPT N-values from 127 tests ranged from 24 to refusal (defined as >100 blows/0.3 m (1 ft)), with a median value of 75 blows/0.3 m (1 ft). These are plotted versus depth on Figure 2.5-226. The N_{60} median value adjusted for individual hammer energy is 100 blows/0.3 m (1 ft). The effective angle of internal friction of a very dense sand (N = 100 blows/0.3 m (1 ft)) would typically be taken as over 40 degrees (SSAR Reference 150). However, with the moderately high silt content, φ' has been limited to 40 degrees with c' = 0. The unit weight of 20.4 kN/m^3 (130 pcf) reflects the very dense nature of the Zone IIB saprolite.

The shear wave velocities measured in the soil by suspension P-S velocity logging and CPT seismic testing during the Unit 3 field investigation are shown in Figure 2.5-227. The average shear wave velocity ranges from about 366 m/s (1200 fps) to 762 m/s (2500 fps) with a best estimate of about 488 m/s (1600 fps). This is presented in more detail in Section 2.5.4.4 and Section 2.5.4.7.

The high strain (i.e., in the range of 0.25 to 0.5 percent) elastic modulus value has been derived using the relationship with SPT N-value given in SSAR Reference 151. The shear modulus value has been obtained from the elastic modulus values using the relationship between elastic modulus, shear modulus and Poisson’s ratio (SSAR Reference 150). The low strain (i.e., 10^{-4} percent) shear modulus has been derived from the best estimate shear wave velocity of 488 m/s (1600 fps).

In Table 2.5-212, the value of unit coefficient of subgrade reaction is based on the value for dense sand provided by Terzaghi (SSAR Reference 152). The earth pressure coefficients are Rankine values, assuming level backfill and a zero friction angle between the soil and the wall (see also Section 2.5.4.10.3).

Structural Fill

Structural fill for placing beneath and around major power block structures is obtained from crushing the sound rock removed from the deep excavation for some of these structures, including the Reactor Building, Fuel Building, Control Building and Radwaste Building. The rock is crushed down to well-graded, angular or sub-angular gravel-sized particles. It is compacted with heavy equipment in thin lifts to a dry density that is at least 95 percent of the maximum dry density obtained from ASTM D 1557 (SSAR Reference 165) (see also Section 2.5.4.5). Based on this, N_{60} = 50 blows/0.3 m (1 ft) and φ' = 40 degrees were selected as reasonable and conservative.

c. **RCTS Testing**

The results of the three RCTS tests are presented in Appendix 2.5.4AAS1 and illustrated in Figure 2.5-223. Two of the tests were on Zone IIA saprolites (each an SM sample, obtained using a Shelby tube) and one test was on a sample of Zone IIB saprolite (also SM, obtained using a rotary Pitcher barrel sampler). The test results on Figure 2.5-223 show normalized shear modulus (G/G_{max}) and material damping ratio, D, versus shear strain, for both the resonant column and torsional shear modes. The results are shown for a confining pressure approximately equal to the in-situ confining pressure.

Comparison of the RCTS results with the generic curves used in the seismic soil column analyses is illustrated and discussed in Section 2.5.4.7.

d. **Electrical Resistivity and Chemical Properties**

When assessing the corrosion potential of soils, electrical resistivity and selected chemical testing results are typically used in combination. Field electrical resistivity and laboratory chemical tests were performed on the Zone IIA and Zone IIB saprolites during the Unit 3 subsurface investigation, and the results of the tests are given in Appendix 2.5.4AA. The results of the chemical tests are also shown in Table 2.5-210. The results are described in the following paragraphs.

Zone IIA Saprolite

The electrical resistivity measured in two arrays ranges from over 100 ohm-m close to the surface to around 500 ohm-m at 9.1 m (30 ft) depth. The chloride content of the soil, measured in 14 tests, ranges from about 2 to 210 parts per million (ppm), with a median value of about

6 ppm. These results suggest very low corrosion potential. The pH, measured in 15 tests, ranges from 4.7 to 7, with a median of 5.8. These pH results indicate a higher corrosion potential than the resistivity or chloride results. The sulfate content measured in 11 tests ranges from about 3 to 11 ppm, indicating that no special sulfate resisting cement is required.

Zone IIB Saprolite

The electrical resistivity measured in two arrays was about 450 ohm-m at 15.2 m (50 ft) depth. The chloride content, measured in 4 tests, is less than 10 ppm, while the pH ranges from 6.7 to 7.4. These results suggest very low corrosion potential. The sulfate content measured in 4 tests ranges from about 2 to 9 ppm, indicating that no special sulfate resisting cement is required.

2.5.4.3 Foundation Interfaces

NAPS ESP COL 2.5-2

The locations of site exploration points for the Unit 3 subsurface investigation, including borings, observation wells, CPTs, electrical resistivity tests, and test pits made in the power block area are shown on Figure 2.5-221. Borings from previous exploration programs are also shown. Exploration points outside the power block area are shown on Figure 2.5-222.

NAPS ESP COL 2.5-3

Figure 2.5-214 shows the excavation plan for the safety-related and other major facilities, and includes the plan outline of these structures. Figure 2.5-214 gives the plan dimensions and the bottom of foundation elevations for the major structures. Also shown in Figure 2.5-214 are the locations of the 6 subsurface profiles shown on Figure 2.5-215 through Figure 2.5-220. The cross sections of the structure foundations and the proposed excavation and backfilling limits are superimposed on Figure 2.5-215 through Figure 2.5-220 to produce Figure 2.5-229 through Figure 2.5-234.

NAPS COL 2.0-29-A

Logs of the core borings, observation wells, CPTs and test pits are in Appendix 2.5.4AA.

2.5.4.4 Geophysical Surveys

The geophysical testing for Unit 3 consisted of field electrical resistivity testing, geophysical down-hole testing, and seismic CPTs.

2.5.4.4.1 Field Electrical Resistivity Testing

Field electrical resistivity testing was conducted along two crossing lines located as shown on Figure 2.5-221. The Wenner four-electrode method was used to perform the tests in accordance with ASTM G 57 (Reference 2.5-210). In this method, four electrodes, two for current and two for voltage, are spaced an equal distance apart and inserted about 0.3 m (1 ft) into the ground. A current is sent through the two outer electrodes and voltage is measured at the two inner electrodes. Electrode spacing (“A” spacing) ranged from 0.9 m (3 ft) to 30.5 m (100 ft). The results of the testing are given in Appendix 2.5.4AA and are described relative to corrosion potential in Section 2.5.4.2.5.d.

2.5.4.4.2 Geophysical Down-Hole Testing

This suite of tests was performed in borings B-901 (91.4 m (300.0 ft) depth), B-907 (61.1 m (200.5 ft) depth) and B-909 (61.5 m (201.9 ft) depth). The tests conducted were natural gamma, three arm caliper, resistivity, spontaneous potential, borehole acoustic televiewer logging, boring deviation, and suspension P-S velocity logging. The results of all of these tests and detailed descriptions of the test methods are in Appendix 2.5.4AA. Plots of the shear and compression wave velocity results versus depth are presented in Section 2.5.4.4.4. The descriptions below are summarized from the more detailed description in Appendix 2.5.4AA.

For all of the tests, all three borings were logged as partially-cased borings, filled with clear water or polymer-based drilling mud, with a 102 mm (4 in) PVC or steel casing placed in the top 12.2 m (40 ft) (B-901 and B-907) or 24.4 m (80 ft) (B-909) of soil above bedrock contact during the measurements in the lower rock portions of the borings. The casing was then removed and measurements were performed in the upper soil portion of the borings. The instrument probe receives control signals from, and sends the digitized receiver signals to, instrumentation on the surface via an armored 4-conductor cable. The cable is wound onto the drum of a winch and is used to support the probe.

a. **Natural Gamma and 3-Arm Caliper**

Natural gamma and caliper data were collected using a Model 3ACS 3-leg caliper probe, manufactured by Robertson Geologging, Ltd. With this tool, caliper measurements were collected concurrently with the measurement of natural gamma emission from the borehole wall. The probe is 2.08 m (6.82 ft) long and 38 mm (1.5 in) in diameter and can:

- Measure boring diameter and volume
- Locate hard and soft formations
- Locate fissures, caving, pinching and casing damage
- Identify bed boundaries
- Correlate strata between borings
- Provide natural gamma measurements

Natural gamma measurements rely upon small quantities of radioactive material contained in all rocks to emit gamma radiation as they decay. The measurement is useful because the radioactive elements are concentrated in certain rock types, e.g., clay or shales, and depleted in others, e.g., sandstone or coal.

For testing, the probe was lowered to the bottom of the boring where the caliper legs were opened, and data collection was begun. The probe was returned to the surface at a rate of 3.0 m (10 ft)/minute, collecting data continuously at 0.015 m (0.05 ft) spacing.

b. **Resistivity, Spontaneous Potential and Natural Gamma**

Resistivity, spontaneous potential, and natural gamma data were collected using a Model ELXG electric log probe, manufactured by Robertson Geologging, Ltd. The probe, which is 2.5 m (8.2 ft) long and 44 mm (1.73 in) in diameter, measures single point resistance, short and long normal resistivity, spontaneous potential, and natural gamma, and can:

- Identify bed boundaries
- Correlate strata between borings
- Identify strata geometry (shale indication)
- Provide natural gamma measurements

For testing, the probe was lowered to the bottom of the boring, and data collection was begun. The probe was returned to the surface at a rate of 3.0 m (10 ft)/minute, collecting data continuously at 0.015 m (0.05 ft) spacing.

c. **Acoustic Televiewer and Borehole Deviation Measurement**

Acoustic image and boring deviation data were collected using a High Resolution Acoustic Televiewer probe, manufactured by Robertson Geologging, Ltd. The probe, which is 2.31 m (7.58 ft) long and 48 mm (1.9 in) in diameter, is fitted with upper and lower four-band centralizers, and can:

- Measure boring inclination and deviation from vertical
- Determine need to correct soil and geophysical log depths to true vertical depths
- Provide acoustic imaging of the borehole to identify fractures, dikes, and weathered zones, and determine dip and azimuth of these features

This system produces images of the borehole wall based on the amplitude and travel time of an ultrasonic beam reflected from the formation wall. The strength of the reflected signal from the formation wall depends primarily upon the impedance contrast between the clear water or drilling fluid and the wall. In the North Anna rock borings, the contrast between the fluid and the rock formation generally provided high contrast. The acoustic wave propagates along the axis of the probe and is then reflected perpendicular to this axis by a reflector that focuses the beam to a 2.5 mm (0.1 in) diameter spot about 50 mm (2 in) from the central axis of the probe. This reflector is able to rotate. During the survey, data were collected at 360 samples per revolution.

The probe contains a fluxgate magnetometer to monitor magnetic north, and all raw televiewer data are referenced to magnetic north. In addition, a 3-axis accelerometer is enclosed in the probe, and boring deviation data are recorded during the logging runs, to permit correction of structure dip angle from apparent dip to true dip in non-vertical borings.

For testing, the probe was lowered to the bottom of the boring, and data collection was begun. The probe was returned to the surface at a rate of 0.91 m (3 ft)/minute, collecting data continuously at 0.0024 m (0.008 ft) intervals. The data were presented on a computer screen for operator review during the logging run, and stored on hard disk for later processing.

d. **Suspension P-S Logger**

Suspension soil and rock velocity measurements were performed using the Robertson Geologging USB Micrologger II digital recorder with a digital OYO Suspension P-S Logging Probe. This system directly determines the average in-situ horizontal shear and compressional wave velocity measurements of a 1.0 m (3.3 ft) high segment of the soil and rock column surrounding the borehole by measuring the elapsed time between arrivals of a wave propagating upwards through the soil and rock column.

Suspension P-S velocity logging uses a 7.0 m (23 ft) long cable suspended probe containing a source near the bottom, and two geophone receivers spaced 1.0 m (3.3 ft) apart. The probe is lowered into the borehole to a specified depth where the source generates a pressure wave in the borehole fluid (drilling mud). The pressure wave is converted to seismic waves (P-wave and S-wave) at the borehole wall. At each receiver location, the P- and S-waves are converted to pressure waves in the fluid and received by the geophones mounted in the probe, which in turn send the data to a recorder on the surface. At each measurement depth, two opposite horizontal records and one vertical record are obtained. This procedure is typically repeated every 0.5 m (1.65 ft) or 1.0 m (3.3 ft) as the probe is moved from the bottom of the borehole towards the ground. The elapsed time between arrivals of the waves at the geophone receivers is used to determine the average velocity of a 1.0 m (3.3 ft) high column of soil or rock around the borehole. For quality assurance, analysis is also performed on source-to-receiver data.

2.5.4.4.3 **Seismic Tests with Cone Penetrometer**

The tests were performed at 1.5 m (5 ft) intervals in C-902, C-916, C-921 and CPT-923. Shear waves were generated by striking a heavy beam adjacent to the CPT location. Only shear waves were generated. The wave arrival was recorded by a geophone attached near the bottom of the cone string. The results of these seismic CPTs are provided in Appendix 2.5.4AA, and discussed in Section 2.5.4.4.4.

2.5.4.4.4 **Results of Shear and Compression Wave Velocity Tests**

a. **Soil**

The measurements of shear wave velocity (V_s) from suspension P-S logging and seismic CPT tests in the Zone IIA and Zone IIB saprolite (and top of Zone III weathered rock) are shown versus depth in Figure 2.5-227. The corresponding measurements of compression wave velocity (V_p), from the suspension P-S logging are shown in

Figure 2.5-235. Low strain Poisson's ratio can be determined from a relationship between V_s and V_p (SSAR Reference 150). A plot of Poisson's ratio versus depth derived from the suspension P-S logging V_s and V_p measurements is shown in Figure 2.5-236. Note that on these plots, the Zone IIA saprolite extends to about 7.6 m (25 ft) depth in boring B-909, and to about 10.7 m (35 ft) depth in borings B-901 and B-907.

For the Zone IIA saprolite, the average shear wave velocity generally increases with depth from around 15.2 m/s (500 fps) at the ground surface to 366 m/s (1200 fps) as it transitions to Zone IIB saprolite. The median value within the layer is about 259 m/s (850 fps). This compares with a median of about 290 m/s (950 fps) noted in the SSAR. The results of the compression wave tests in Zone IIA saprolite are fairly consistent at around 549 m/s (1800 fps), while the low strain Poisson's ratio can be taken as 0.35.

For the Zone IIB saprolite, the average shear wave velocity generally ranges from around 366 m/s (1200 fps) to 762 m/s (2500 fps) as it transitions to Zone III saprolite. The median value within the layer is about 488 m/s (1600 fps) which is the same as noted in the SSAR. The results of the compression wave tests in Zone IIB saprolite in Figure 2.5-235 reflect the compression velocity of water. The compression wave velocity from SSAR Table 2.5-45 of 1067 m/s (3500 fps) was used, with a low strain Poisson's ratio of 0.37.

b. **Rock**

Figure 2.5-237 shows the measurements of V_s from suspension P-S logging in the Zone III, Zone III-IV and Zone IV bedrock versus elevation. Figure 2.5-238 shows the corresponding measurements of V_p, while Figure 2.5-239 shows Poisson's ratio versus elevation derived from V_s and V_p. These measurements were taken in the power block area, i.e., at the Reactor Building, at the Fuel Building, and close to the FWSC. The elevations of the bottom of the RB/FB building mat (Elevation 68.3 m (224 ft)), and Control Building mat (Elevation 73.5 m (241 ft)) are shown on these figures as well as the top of competent material in this area (top of Zone III-IV at about Elevation 83.2 m (273 ft)), and the design plant grade (Elevation 88.4 m (290 ft)).

Based on a review of the V_s versus elevation information in Figure 2.5-237, and the RQD data in Figure 2.5-224 as described in Section 2.5.4.2.5.a, it was concluded that the overall shear wave velocities of the rock as defined by the three rock zones (III, III-IV and IV) are somewhat higher at the Unit 3 plant location than described in the SSAR. For Zone III weathered rock, the range of V_s is approximately 610 m/s (2000 fps) to 1219 m/s (4000 fps), with a best estimate value of 914 m/s (3000 fps). For Zone III-IV partially weathered rock, the range of V_s is approximately 914 m/s (3000 fps) to 2438 m/s (8000 fps), with a best estimate value of 1372 m/s (4500 fps). For Zone IV fresh rock, the range of V_s is approximately 2438 m/s (8,000 fps) to 3048 m/s (10,000 fps), with a best estimate value of 2743 m/s (9000 fps).

In Figure 2.5-237, Zone IV bedrock extends up to around Elevation 61 m (200 ft), although about 6.1 m (20 ft) of Zone III rock was identified (from the V_s, RQD and core description) as extending below Elevation 61.0 m (200 ft) in B-901. From Elevation 61.0 m (200 ft) to about Elevation 68.6 m (225 ft), all the borings show Zone III-IV. Above about Elevation 68.6 m (225 ft), B-907 shows mostly Zone III material while B-901 shows Zone III-IV rock. In B-909, rock was not encountered above about Elevation 68.6 m (225 ft). These V_s profiles demonstrate that, whereas previously the “top of competent rock” was the top of the Zone III-IV, the shear wave velocities in the Zone III rock can be high enough (e.g., in B-907) that, in some instances, Zone III can be included in the “competent rock” description. As noted above, top of competent rock at the location of the RB and FB is at about Elevation 83.2 m (273 ft). The V_s profiles also demonstrate, along with the RQD profile in Figure 2.5-224, that above about Elevation 53.3 m (175 ft), weathered/fractured zones can be encountered; however, there is no pattern to where these zones occur, indicating the randomized process of weathering.

2.5.4.5 Excavation and Backfill

NAPS ESP COL 2.5-3

This section describes the following topics:

- The extent (horizontally and vertically) of Seismic Category I excavations, fills and slopes
- Excavation methods and stability
- Backfill sources, quantities, compaction specifications and quality control

2.5.4.5.1 Extent of Excavations, Fills and Slopes

Figure 2.5-214, the bottom of foundation plan, shows the extent of excavations, fills and slopes for Unit 3. These are shown in cross-section in Figure 2.5-229 through Figure 2.5-234. To obtain the design plant grade of Elevation 88.4 m (290 ft), up to 12.2 m (40 ft) of soil will be excavated. The location of original ground surface is shown in the cross-sections. There are some lower areas to the northeast that will be backfilled. (Directions are with respect to true north.) The total estimated cut to achieve finish grade is about 550,500 m^3 (720,000 cubic yards), while the amount of backfilling is about 336,400 m^3 (440,000 cubic yards). Benched 3-horizontal to 1-vertical (3H:1V) slopes extend up from plant grade around the southern perimeter of the area. On the northeastern perimeter of plant grade, a 2 percent slope extends downwards towards the plant grade for Units 1 and 2. The stability of the 3H:1V slopes is addressed in Section 2.5.5.

Figure 2.5-214 shows the outline of the power block foundations. The vertical cuts in soil shown on the foundation cross-sections in Figure 2.5-229 through Figure 2.5-234 will be supported by a tied-back wall system, with the tie-backs anchored into the underlying bedrock where feasible.

2.5.4.5.2 Excavation Methods and Stability

a. **Excavation in Soil**

Excavation in the soils (Zones IIA and IIB) and any existing fills is achieved with conventional excavating equipment. Excavation of less than 6.1 m (20 ft) in height will adhere to OSHA regulations (SSAR Reference 162). As noted in the previous section, a vertical soil cut and tie-back system will be used to support the power block excavation. The slopes around the perimeter of the power block area are no steeper than 3H to 1V, with benches every 6.1 m (20 ft) of height. Since the saprolitic soils can be highly erosive, even temporary slopes cut into the saprolite are sealed and protected.

b. **Excavation in Rock**

Excavation in the Zone III moderately to severely weathered rock is achieved using conventional earthmoving equipment. A vertical soil cut and tie-back system will be used to support the excavation, where necessary.

Excavation made for the originally planned Units 3 and 4 in the slightly to moderately weathered rock (Zone III-IV) and fresh to slightly weathered rock (Zone IV) is documented in SSAR Reference 163. Techniques employed were similar to those used for Units 1 and 2 (SSAR Reference 164) but with "lessons learned" applied. The methods of rock excavation outlined below for Unit 3 are based, in part, on the methods that worked successfully for Units 1 and 2 and the originally planned Units 3 and 4. Unit 3 is approximately 460 m (1500 ft) from the center of the Unit 2 containment building, whereas the originally planned Unit 3 Reactor Building was only about 90 m (300 ft) from the Unit 2 Reactor Building. Thus, the following techniques to reduce vibrations that worked for the originally planned Unit 3 will be used and will be effective for the new Unit 3:

- Controlled blasting techniques, including cushion blasting, pre-splitting and line drilling may be used, with appropriately dimensioned bench lifts. The blasted faces are vertical except where the foliation dip is into the excavation. There, the excavation may be parallel to the foliation dip (typically about 1-H to 1-V).
- Any blasting is strictly controlled to preserve the integrity of the rock outside the excavations and to prevent damage to existing structures, equipment, and freshly poured concrete. Peak particle velocity is measured and kept within specified limits that are a function of distance from the blast.
- The rock is reinforced to ensure adequate support and safety. Reinforcing includes installation of rock bolts in finished rock faces (typically at around 1.5 m (5 ft) centers), and the use of welded wire mesh. Necessary measures are taken when weathered or fractured zones are encountered. Instrumentation such as slope indicators and extensometers are installed to monitor rock movements, especially on the foliation dip slopes.

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- The excavation for safety-related structures will be geologically mapped and photographed by experienced geologists. Unforeseen geologic features that are encountered will be evaluated. The NRC will be notified no later than 30 days before any excavations for safety-related structures are open to allow for NRC staff examination and evaluation.
- There is no measurable rebound or heave of the sound rock subgrade, and monitoring is not needed.

2.5.4.5.3 Structural Fill Sources, Compaction and Quality Control

Although a large amount of Zone IIA soil will be excavated for Unit 3, this material will not be used as structural fill to support Seismic Category I or II structures.

Structural fill is either lean concrete or a sound, well-graded granular material. The anticipated extent of the concrete and granular fill is shown on the foundation cross-sections on Figure 2.5-229 through Figure 2.5-234. The concrete fill is used to replace any moderately to severely weathered rock (Zone III) exposed at the bottom of the excavations for the Seismic Category I RB/FB and Control Building foundation mats. The concrete fill will be designed to result in a shear wave velocity in the same range as that of the Zone III-IV rock.

The granular structural fill material does not exist naturally on site. However, given the large amount of rock that will need to be excavated for Unit 3, it will be economical to set up a crushing and blending plant onsite to produce crushed aggregate to the required gradation specifications for use as structural fill. The rock will be crushed down to well-graded, angular or sub-angular gravel-sized particles, with less than 5 percent passing the number 200 sieve. The soundness of the aggregate will be confirmed using sulfate soundness and Los Angeles abrasion tests. This structural fill will be placed in thin lifts and compacted to at least 95 percent of the maximum dry density as determined by ASTM D 1557 (SSAR Reference 165), and to within 3 percent of its optimum moisture content. Compaction will be performed with a heavy steel-drummed vibratory roller, except within 1.5 m (5 ft) of a structure wall, where smaller compaction equipment will be used in conjunction with reduced lift thickness to minimize excess pressures against the wall. As noted in Section 2.5.4.2.5.b, based on the type of material and its degree of compaction, N_{60} = 50 blows/0.3 m (1 ft) and φ' = 40 degrees were assumed as reasonable and conservative for this structural fill.

As an alternative or supplement to the onsite crushed rock, dense-graded aggregate can be used as structural fill material. Dense graded aggregate such as Size 21A or 21B as specified by the Virginia Department of Transportation Road and Bridge Specifications (SSAR Reference 166) is suitable material.

Fill placement and compaction control procedures will be addressed in a technical specification that includes requirements for suitable fill, sufficient testing to address potential material variations, and in-place density testing frequency, i.e., a minimum of one test per 930 m^2 (10,000 ft^2) of fill placed. It also includes requirements for an on-site testing laboratory for quality control (gradation, moisture-density, placement, compaction, etc.) and requirements to ensure that the fill operations conform to the earthwork specification. The soil testing firm is required to be independent of the earthwork contractor and to have an approved quality assurance program. Sufficient laboratory compaction (modified Proctor) and grain size distribution tests will be performed to ensure that variations in the fill material are accounted for. (Variations in the crushed and blended rock are expected to be minimal.)

A test fill program is also included for the purposes of determining an optimum size of roller, number of passes, lift thickness, and other relevant data for achievement of the specified compaction.

2.5.4.5.4 Control of Groundwater During Excavation

Construction dewatering is presented in Section 2.5.4.6.2. Since the saprolitic soils can be highly erosive, sumps and ditches constructed for dewatering are lined. The tops of excavations are sloped back to prevent runoff down the excavated slopes during heavy rainfall.

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2.5.4.6 Groundwater Conditions

2.5.4.6.1 Groundwater Measurements and Elevations

Groundwater is present in unconfined conditions in both the surficial sediments and underlying bedrock at the Unit 3 site. Seven observation wells installed for the Unit 3 investigation (along with nine wells installed at the site as part of the ESP subsurface investigation program) have exhibited groundwater levels ranging from about Elevation 72.5 m (238 ft) to Elevation 95.7 m (314 ft) between December 2002 and August 2007. (The groundwater generally occurs at depths ranging from about 5.5 m (18 ft) to 7.6 m (25 ft) below the present-day ground surface in the main Unit 3 power block area.)

The logs and details of these seven wells, and tests in the wells, are given in Appendix 2.5.4AA. Details of measured groundwater levels and their fluctuations are given in Section 2.4.12. Hydraulic conductivity values for the saprolite based on slug tests performed in eleven of the observation wells range from 0.076 m (0.25 ft) to 3.02 m (9.9 ft)/day, with a geometric mean value of 0.53 m (1.74 ft)/day. The hydraulic conductivity of the underlying shallow bedrock as determined from slug tests performed in two of the wells and packer tests performed in one of the wells is estimated to range from about 0.15 m (0.5 ft) to 1.92 m (6.3 ft)/day, with a geometric mean value of 0.62 m (2.05 ft)/day. Groundwater movement at the site is generally to the north and east, toward Lake Anna. A detailed description of groundwater conditions is provided in Section 2.4.12.

Groundwater levels at the site require temporary dewatering of foundation excavations extending below the water table during construction of Unit 3. This construction dewatering is performed in a manner that minimizes drawdown effects on the surrounding environment. Drawdown effects are expected to be limited to the NAPS site. The relatively low permeability of the saprolite and underlying rock means that sumps and pumps should be sufficient for successful construction dewatering, as presented in Section 2.5.4.6.2.

The maximum allowable ground water level for operation of the power block area of Unit 3 is Elevation 87.8 m (288 ft) which is at 0.6 m (2 ft) below design plant grade at Elevation 88.4 m (290 ft). Section 2.4.12.4 indicates that the maximum groundwater level in the power block area of Unit 3 is Elevation 86.3 m (283 ft).

2.5.4.6.2 Construction Dewatering and Seepage

Dewatering for all major excavations is achieved by gravity-type systems.

a. **Soils**

Due to the relatively impermeable nature of even the coarse-grained saprolite, sump-pumping of ditches is adequate to dewater the soil. These ditches are advanced below the progressing excavation grade.

During the construction of Units 1 and 2 and originally planned Units 3 and 4, plant excavation and dewatering was significant in causing local groundwater levels to decline. However, the extent of the area of influence of the construction dewatering was estimated to be a radius of less than 152 m (500 ft) due to the low permeability of the materials being dewatered (SSAR Reference 164).

b. **Rock**

Sump-pumping is used to collect water from relief drains that are installed in the major rock excavation walls to prevent hydrostatic pressure buildup behind the walls. Such relief wells were spaced on 6.1 m (20 ft) centers around the perimeters of the originally planned Units 3 and 4 containment excavations.

Although an approximately 12.2 m (40 ft) head existed between excavation grade and the North Anna Reservoir during the final stages of excavation for the originally planned Units 3 and 4, no dewatering difficulties were encountered, due to the tight nature of the joints in the rock below about Elevation 73.2 m (240 ft). The excavation for Unit 3 is at least 305 m (1000 ft) from Lake Anna, and so negligible seepage effects from the lake are anticipated.

2.5.4.6.3 Effect of Groundwater Conditions on Foundation Stability

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Maximum allowable groundwater level is at least 0.6 m (2 ft) below plant grade, i.e., Elevation 87.8 m (288 ft). This water level was used in bearing capacity and settlement analyses and in computing hydrostatic pressures on the buried structure walls (Section 2.5.4.10). As described in Section 2.5.4.10, there are no buoyancy issues with deep buried structures because of the appreciable dead loads imposed by these structures. Large diameter buried piping such as the circulating water pipes are designed to resist buoyancy when empty.

No permanent dewatering system is required for Unit 3.

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2.5.4.7 Response of Soil and Rock to Dynamic Loading

The RB/FB common basemat at Unit 3 is founded on Zone III-IV or Zone IV bedrock or on concrete placed on Zone III-IV or Zone IV bedrock. A similar scheme is followed for the CB foundation, although some thin layers of Zone III material may be present at foundation level. The other Seismic Category I structure (the FWSC) is founded on compacted structural fill placed on top of Zone III weathered rock. (The structural fill replaces in-situ saprolite.) The foregoing foundation subgrades are illustrated on Figure 2.5-229 through Figure 2.5-234.

The seismic acceleration at the sound bedrock level is amplified or attenuated up through the weathered rock and soil column. To estimate this amplification or attenuation, the following data are required:

- Shear wave velocity profiles of the rock and soil overlying hard rock
- Variation with strain of the shear modulus and damping values of the weathered rock and soil
- Site-specific seismic acceleration-time histories

2.5.4.7.1 Shear Wave Velocity Profile

NAPS ESP COL 2.5-9

Various measurements were made at the Unit 3 site to obtain estimates of the shear wave velocity in the soil and rock. These are summarized in Section 2.5.4.4. The materials of interest here are the Zone IIA and Zone IIB saprolitic soils, the structural fill, the Zone III weathered rock, the Zone III-IV slightly to moderately weathered rock, and the Zone IV slightly weathered to fresh rock. Since the bedrock supports the majority of the Seismic Category I structures, it is considered first.

a. **Bedrock**

Shear wave velocity of the bedrock at the RB/FB basemat (B-901 and B-907) and the edge of the CB (B-909) is shown versus elevation in Figure 2.5-237. Below about Elevation 44.2 m (145 ft), the shear wave velocity is fairly constant at between around 2740 m/s (9,000 fps) and 3050 m/s (10,000 fps). As noted in Section 2.5.4.4.4, Figure 2.5-237 shows Zone IV bedrock extending up to around Elevation 61 m (200 ft), although about 6.1 m (20 ft) of Zone III rock was identified (from V_s, RQD and core description) extending below Elevation 61.0 m (200 ft) in B-901. From Elevation 61.0 m (200 ft) to about Elevation 68.6 m (225 ft), all the borings show Zone III-IV with shear wave velocities ranging from about 1220 m/s (4000 fps) to 2440 m/s (8000 fps). Above about Elevation 68.6 m (225 ft), B-907 shows mostly Zone III material while B-901 shows Zone III-IV rock, with top of competent material (mostly Zone III-IV rock but can include Zone III) at Elevation 83.2 m (273 ft).

Figure 2.5-240 shows best-fit values applied to the measured shear wave velocity profiles in Figure 2.5-237. Above about Elevation 56.1 m (184 ft), there are two profiles, with one representing the mostly unweathered and unfractured rock profile, and the other the more weathered and fractured profile. The median shear wave velocities derived from the Figure 2.5-237 values and used in the randomization model for input into the SHAKE (Reference 2.5-211) analysis (Section 2.5.4.7.4) are shown in Figure 2.5-241. The median profile indicates that V_s = 2800 m/s (9200 fps) is reached at about Elevation 45.1 m (148 ft). Figure 2.5-242 shows the 60 randomized rock profiles used in the SHAKE analysis, with these profiles enveloping the two design profiles.

Table 2.0-201 provides an evaluation of DCD site parameter values and corresponding Unit 3 site characteristic values for shear wave velocity.

b. **Soil**

Two soil profiles were considered for SHAKE analysis. The first is a natural soil profile that is outside the power block since all of the natural soil is removed from within the power block area. The profile is in the vicinity of boring B-947, on the planned 3H:1V slope to the southeast of the FWSC shown on Figure 2.5-214, with ground elevation at around 96.0 m (315 ft). Boring B-947 is shown on Subsurface Profile D-D' on Figure 2.5-218. This profile was used in the slope stability analyses presented in Section 2.5.5 and for the peak ground acceleration used in the liquefaction analysis in Section 2.5.4.8.

The second soil profile is that of the engineered structural fill beneath the FWSC. As noted in Section 2.5.4.5.3, the primary source of structural fill is crushed rock obtained from the power block excavation.

For the natural soil profile, the measured shear wave velocity profiles in Figure 2.5-227 were averaged vertically in 1.5 m (5 ft) intervals to obtain the average, upper bound and lower bound profiles shown in Figure 2.5-243. As with the bedrock profile, this soil profile was randomized for input into the SHAKE analysis. At the natural soil profile location, subsurface information indicated that the top of competent rock was at about Elevation 76.2 m (250 ft). The same bedrock profile described above in Section 2.5.4.7.1.a, with top of competent rock at Elevation 83.2 m (273 ft) at the RB location, was assumed for the SHAKE analysis to extend below Elevation 76.2 m (250 ft). (The top of competent material varies in elevation throughout the site, frequently, but not consistently following the changes in original topography of the site. As indicated earlier, Zone III-IV rock is always considered competent, but some Zone III weathered rock is also considered competent.)

For the structural fill beneath the FWSC, there are no measured shear wave velocities, since the fill will be crushed rock obtained from the new plant excavation. To obtain a shear wave velocity profile range, the SPT N-value selected in Section 2.5.4.2.5.b for the fill, i.e., N_{60} = 50 blows/0.3 m (1 ft), was used. Relationships between N-value (adjusted for overburden pressure) and shear wave velocity developed by Seed, et al. (Reference 2.5-212) and Imai and Tonoucchi (Reference 2.5-213) were used to obtain a profile of shear wave velocity versus depth, as shown in Figure 2.5-244. This profile was averaged vertically in 1.5 m (5 ft) intervals to obtain the average shear wave velocity profile shown in Figure 2.5-245. As shown in Figure 2.5-232, the top of weathered rock beneath the FWSC is at around Elevation 73.2 m (240 ft), overlain by Zone IIB saprolite. For the dynamic analysis, it was conservatively assumed that the Zone IIB saprolite is removed and structural fill placed above about Elevation 73.2 m (240 ft) to the bottom of the FWSC at Elevation 86.0 m (282 ft), as illustrated in Figure 2.5-245. The upper and lower bounds shown in this figure are 1.225 and 0.816 times the mean value of shear wave velocity, respectively, which correspond to 1.5 and 0.67 times the shear modulus. As with the bedrock profile, this soil fill profile was randomized for input into the SHAKE analysis. As noted above, subsurface information indicated that the top of weathered rock was at about Elevation 73.2 m (240 ft). The very high SPT N-values at the bottom of the boring beneath the FWSC (B-921) suggest that the top of weathered rock in this case can be assumed to be the top of competent material. The same bedrock profile described above in Section 2.5.4.7.1.a, with top of competent rock at Elevation 83.2 m (273 ft) at the RB location, was assumed for the SHAKE analysis to extend below Elevation 73.2 m (240 ft) at the FWSC. Table 2.0-201 provides an evaluation of the DCD site parameter value and the corresponding Unit 3 site characteristic value for shear wave velocity.

NAPS COL 2.0-29-A

2.5.4.7.2 Variation of Shear Modulus and Damping with Strain

a. **Shear Modulus**

The shear modulus reduction curve for the Zone IIA saprolite is the same as used for the Zone IIA saprolite in the SSAR, i.e., Curve 1 in SSAR Figure 2.5-63. This curve is reproduced here in Figure 2.5-246, labeled “Recommended for Natural Soil.” A series of grain size tests on the Zone IIB saprolite indicated that all of the samples tested were sands, with no appreciable gravel content. Thus, Curve 1 in SSAR Figure 2.5-63 was also used for the Zone IIB saprolite, and labeled “Recommended for Natural Soil” in Figure 2.5-246. The typical thickness of the saprolite is about 10.7 m (35 ft). Curve 1 is almost identical to the average of the EPRI curves (SSAR Reference 170) for depths 0 to 6.1 m (20 ft), and 6.7 m (20 ft) to 15.2 m (50 ft).

The results of the RCTS tests (normalized shear modulus (G/G_{max}) versus shear strain) from Figure 2.5-223 are superimposed on Curve 1 in Figure 2.5-247. These results show good agreement with Curve 1, and so no additional SHAKE runs were made using the RCTS shear modulus reduction curves. Note that the median thickness of the Zone IIA saprolite encountered in the Unit 3 borings was about 7.6 m (25 ft), and approximately 80 percent of the material was classified as silty sand (SM). The two silty sand samples of Zone IIA saprolite tested in RCTS are thus considered sufficient and representative. Similarly, the median thickness of the Zone IIB saprolite encountered in the Unit 3 borings was about 2.7 m (9 ft), and all of this material was classified as silty sand (SM). Thus the sample of Zone IIB silty sand tested in RCTS is considered sufficient and representative.

As noted in Section 2.5.4.2.5.b, the primary source of structural fill is bedrock excavated to construct the Unit 3 power block, crushed down to well-graded, angular gravel-sized particles. Curve 2 in SSAR Figure 2.5-63, which was derived for a gravel-type material, was selected as the shear modulus reduction curve for this structural fill and is included in Figure 2.5-246. Curve 3 in SSAR Figure 2.5-63 was used for the Zone III weathered rock. The shear modulus of the Zone IV and Zone III-IV weathered rock was considered non-strain dependent.

b. **Damping**

The typical thickness of the saprolite and the structural fill is about 10.7 m (35 ft). For the granular materials (Zone IIA and Zone IIB saprolite, and the structural fill), the average of the EPRI curves (SSAR Reference 170) for depths 0 to 6.1 m (20 ft), and 6.1 m (20 ft) to 15.2 m (50 ft) was selected. This curve is shown on Figure 2.5-248. Curve 3 in SSAR Figure 2.5-64 is used for the Zone III weathered rock. This curve is also shown on Figure 2.5-248.

Figure 2.5-247 shows the results of the RCTS tests from Figure 2.5-223 for material damping ratio D versus shear strain superimposed on the granular soils curve from Figure 2.5-248. These results show reasonable agreement, and so no additional SHAKE runs were made using the RCTS damping ratio reduction curves.

There is no variation of damping ratio of the Zone III-IV or Zone IV rock with cyclic shear strain. However, this rock has some intrinsic damping properties. A value of 1 percent was selected for the damping ratio.

2.5.4.7.3 Site Specific Acceleration-Time Histories

The time histories for the Unit 3 site are described in SSAR Section 2.5.4.7.3. These time histories were used for the rock and soil column amplification/attenuation analysis described in Section 2.5.4.7.4.

2.5.4.7.4 Rock and Soil Column Amplification/Attenuation Analysis

NAPS ESP COL 2.5-5

The SHAKE2000 (Reference 2.5-211) computer program was used to compute the site dynamic responses for the soil and rock profiles described in Section 2.5.4.7.1 and the variation of shear modulus and damping ratio with strain described in Section 2.5.4.7.2. The analysis used the acceleration-time histories described in Section 2.5.4.7.3. For the low frequency case, an earthquake with moment magnitude of 7.2 and an acceleration at hard bedrock level ($V_s \geq$ 2800 m/s (9200 fps)) of 0.15g was used in the SHAKE2000 analysis, while for the high frequency case, an earthquake with moment magnitude of 5.4 and an acceleration at hard bedrock level of 0.39g was used. One rock profile and two soil profiles were analyzed.

a. **Rock**

Figure 2.5-242 shows the 60 randomized rock profiles used in the SHAKE analysis to obtain the seismic response at the top of competent material, which is at Elevation 83.2 m (273 ft) at the RB/FB location. The response spectrum at the top of competent material is shown in Figure 2.5-205. Response spectra at the horizons that represent the bottom of the RB/FB basemat and the bottom of the CB basemat were also developed from the SHAKE runs. These are shown in Figure 2.5-206 for the CB and in Figure 2.5-207 for the RB/FB.

b. **Soil**

For the natural soil profile, the randomized profile described in Section 2.5.4.7.1 along with the shear modulus and damping ratio relationships with strain described in Section 2.5.4.7.2 were input into the SHAKE analysis. Figure 2.5-249 and Figure 2.5-250 show the maximum acceleration versus depth profiles obtained from SHAKE for the low and high frequency earthquakes, respectively. The mean values on these profiles are used as input into the slope stability analyses described in Section 2.5.5. The mean peak ground acceleration is used as input into the liquefaction analysis for the Unit 3 soils described in Section 2.5.4.8. The peak acceleration at the natural ground surface using the low frequency earthquake is 0.30g, while the corresponding acceleration using the high frequency earthquake is 0.56g.

For the structural fill profile, the randomized profile described in Section 2.5.4.7.1 along with the shear modulus and damping ratio relationships with strain described in Section 2.5.4.7.2 were input into the SHAKE analysis. The seismic response spectrum developed at the top of the fill column corresponds to that for use in the FWSC design, as shown in Figure 2.5-208.

NAPS COL 2.0-29-A

2.5.4.8 Liquefaction Potential

The Zone IIB saprolitic soils are extremely dense and the Zone III weathered rock has over 50 percent core stone and has typically been sampled by rock coring. Neither of these materials has liquefaction potential. The primary source of structural fill is bedrock excavated for the Unit 3 power block. This is crushed to angular or sub-angular gravel-sized particles and compacted in thin lifts with a heavy vibratory steel-drummed roller. This fill is not liquefiable. The only material analyzed here regarding liquefaction is the Zone IIA saprolitic soil.

NAPS ESP PC 3.E(7)

The only Seismic Category I structure not founded on rock or on concrete on rock at the Unit 3 site is the FWSC. The FWSC is founded on engineered structural fill after removal of the Zone IIA saprolite. (As described in Section 2.5.4.10, the Zone IIA saprolite has relatively high resistance to bearing failure but can produce excessive settlements under certain conditions. Thus, the Zone IIA saprolite is not used to support Seismic Category I structures, regardless of whether it is potentially liquefiable or not.) No Zone IIA saprolite is within the zone of influence of the FWSC loading. Thus, even if the Zone IIA saprolite is liquefiable, such liquefaction does not impact the stability of any Seismic Category I structure. Note that the Seismic Category II Service Building and the radwaste building are also founded on engineered structural fill.

The peak ground accelerations obtained from the Unit 3 SHAKE analyses through the natural soil profile are less than those reported in the SSAR, due to some slightly different rock and soil profiles, and the randomization process applied to these profiles. The previous liquefaction analyses are described in light of these lower accelerations in Section 2.5.4.8.1. Section 2.5.4.8.1 also contains the results of liquefaction analyses performed on Zone IIA saprolites outside the power block area, based on borings and CPTs performed for Unit 3 outside the perimeter of the vertical soil cut, i.e., analyses of soils that will not be excavated.

2.5.4.8.1 Liquefaction Analyses Performed for Unit 3

This section was developed in accordance with, and conforms to guidance in RG 1.198 (Reference 2.5-214).

a. **Magnitude and Acceleration Values for Unit 3 Liquefaction Analyses**

As noted in Section 2.5.4.7.4, the peak acceleration at the natural ground surface using the low frequency earthquake is 0.30g, while the corresponding acceleration using the high frequency earthquake is 0.56g. The low frequency earthquake had a magnitude of 7.2 and the high frequency earthquake had a magnitude of 5.4.

The 0.30g value was conservatively rounded up to 0.31g for the liquefaction analysis. The 0.31g and 0.56g values, with corresponding magnitudes, were used as the peak ground accelerations for the liquefaction analyses described in the following paragraphs.

As in the SSAR, an acceptable factor of safety (FS) of 1.1 or higher is used in the analyses.

b. **Updated Seismic Margin Assessment**

The seismic margin assessment described in the SSAR for the Units 1 and 2 power block area was modified in the Unit 3 evaluation, maintaining the same assumptions as used in the original study but substituting the Unit 3 design accelerations and moment magnitudes. Magnitude scaling factors of 1.13 and 2.5 were used in the analysis for the low and high frequency earthquakes, respectively. The resulting FS values ranged from about 1.05 to 2.95, with an overall average value of about 1.6.

c. **Analysis of SSAR Samples and CPT Results**

The analysis followed the method proposed by Youd, et al. (SSAR Reference 178). Magnitude scaling factors of 1.13 and 2.5 were used in the analysis for the low and high frequency earthquakes, respectively. The K_σ factor for high overburden pressures was incorporated into the analysis, using a relative density of 60 percent.

Using the magnitude scaling factors for the low and high frequency earthquakes described above, and the Unit 3 peak ground accelerations, the analysis of the SPT results from the SSAR gave FS values against liquefaction greater than 1.1 for those samples that were liquefiable. For the eight CPTs performed, the liquefaction analysis showed a 1.2 m (4 ft) thick zone in one CPT, a 0.61 m (2 ft) thick and a 0.30 m (1 ft) thick zone in one CPT, and two 0.15 m (0.5 ft) thick zones in one CPT where the FS against liquefaction was less than 1.1.

d. **Analysis of Unit 3 SPT Samples and CPT Results**

As noted earlier, at the locations of the majority of the borings and CPTs in the power block area that contains the Seismic Category I structures, the Zone IIA saprolite will be excavated. Thus, analyzing the liquefaction potential of these soils prior to excavation is of little relevance. In this area, there are 18 borings and 9 CPTs that are outside the vertical cut excavation zone and that indicate the presence of Zone IIA saprolite.

Liquefaction analysis of each sample of Zone IIA saprolite obtained by SPT sampling in the 18 borings was performed to determine the FS against liquefaction. The results from the 9 CPTs were also analyzed. The analysis conservatively ignored the age, overconsolidation, and mineralogy/fabric effects of the saprolite. (The saprolite is estimated to be between 0.8 and 1.6 million years old, according to SSAR Reference 176.) Cohesive samples and/or samples above the groundwater table were considered non-susceptible to liquefaction.

The analysis followed the method proposed by Youd, et al. (SSAR Reference 178). This state-of-the-art liquefaction methodology is based on the evolution of the Seed and Idriss "Simplified Procedure" over the past 25 years. Magnitude scaling factors of 1.13 and 2.5 were used in the analysis for the moment magnitude 7.2 (low frequency) and 5.4 (high frequency) earthquakes, respectively. The K_{σ} factor for high overburden pressures was incorporated into the analysis, using a relative density of 60 percent.

The analysis of the SPT results from the 18 borings gave FS values against liquefaction greater than 1.1 for those samples that were liquefiable, except for two samples. For the 9 CPTs analyzed, the liquefaction analysis showed the FS against liquefaction was less than 1.1 in three of them. However, the low FS values occurred mainly in 0.15 m (0.5 ft) or 0.30 m (1.0 ft) thick layers, with the thickest continuous zone of FS < 1.1 being only 0.45 m (1.5 ft) thick.

Using the method outlined in Tokimatsu and Seed (SSAR Reference 179), the maximum estimated dynamic settlement of the Zone IIA saprolite due to earthquake shaking was about 41 mm (1.6 in).

2.5.4.8.2 Conclusions about Liquefaction

Only the Zone IIA saprolites fall into the gradation and relative density categories where liquefaction would be considered possible.

Any liquefaction of the Zone IIA saprolite will not impact the stability of any Seismic Category I or II structure.

The conclusions from the foregoing sections on the analysis of liquefaction potential of Zone IIA saprolite are as follows:

- A seismic margin liquefaction analysis of the Units 1 and 2 power block area was modified to use the Unit 3 seismic parameters (M = 7.2 with 0.31g peak ground acceleration for low frequency and M = 5.4 with 0.56g peak ground acceleration for high frequency) and ignored age, structure, fabric, and mineralogy effects. The analysis gave FS values that were, with very few exceptions, greater than 1.1.
- A state-of-the-art liquefaction analysis of the ESP SPT samples using the low and high frequency Unit 3 seismic parameters gave FS values greater than 1.1 for all the SPT results analyzed. For the ESP CPT measurements, there was a 0.61 m (2 ft) thick and a 1.2 m (4 ft) thick zone where the FS against liquefaction was less than 1.1.
- A state-of-the-art liquefaction analysis of the Unit 3 SPT measurements in borings outside the vertical cut area to be excavated gave FS values against liquefaction greater than 1.1 for those samples that were liquefiable, except for two samples.
- A state-of-the-art liquefaction analysis of the Unit 3 CPT measurements showed the maximum thickness where the FS against liquefaction was less than 1.1, was only 0.45 m (1.5 ft).
- Estimated maximum dynamic settlements of the Zone IIA saprolite due to earthquake shaking are about 41 mm (1.6 in). This settlement will be outside the zone of loading influence of any of the seismic Category I or II structures.

Based on the above analysis results, it can be concluded that a very small percentage of the Zone IIA saprolitic soils have a potential for liquefaction based on the low and high frequency Unit 3 seismic characteristics. The liquefaction analysis did not take into account the beneficial effects of age, structure, fabric, and mineralogy, and thus the chances of any liquefaction occurring are extremely low. Any liquefaction of the Zone IIA saprolite that does occur will not impact the stability of any Unit 3 Seismic Category I or II structure.

2.5.4.9 Earthquake Design Basis

See Sections 2.5.2.6.7 and 2.5.2.7 for the SSE and OBE, respectively.

2.5.4.10 Static Stability

NAPS ESP COL 2.5-6

As with the Units 1 and 2, and the originally planned Units 3 and 4, the Unit 3 RB/FB is founded on Zone III-IV or Zone IV bedrock. If Zone III weathered rock or fractured rock is encountered at foundation subgrade level, then it will be removed and replaced with lean concrete. The subgrade of the other Seismic Category I structures and the Seismic Category II structures depends on their elevation and location. Table 2.5-213 shows the bottom of foundation elevations and depths for the Seismic Category I structures (RB/FB, CB, FWSC), the Seismic Category II structures (Service Building and Ancillary Diesel Building), Turbine Building, and the Radwaste Building. The cross-sections in Figure 2.5-229 through Figure 2.5-234 show the materials supporting these structures (except for the service building). The subsurface profiles beneath the Seismic Category I structures used for bearing capacity and settlement analyses are shown on Figure 2.5-251. The corresponding profiles beneath the Seismic Category II structures and the radwaste building are shown on Figure 2.5-252. There may be several materials immediately beneath the foundations of the larger structures (e.g., the turbine building) because of the variable stratigraphy and the different depths of the parts of the building, and because any Zone IIA saprolite beneath the shallow Seismic Category I or II structures (and the radwaste building) is removed and replaced with structural fill. Table 2.5-213 also shows the design static and dynamic design loads for these structures.

2.5.4.10.1 Bearing Capacity

a. **Bedrock**

The allowable static bearing capacity values for each bedrock zone are given in Table 2.5-214. The Zone III allowable static bearing capacity of 958 kPa (20 ksf) is less than the value of 20 percent of the ultimate crushing strength (or unconfined compressive strength) given in several building codes (SSAR Reference 181). The ultimate crushing strength is given as 6.9 MPa (1.0 kips per square inch (ksi) (144 ksf)) in Table 2.5-212. The 958 kPa (20 ksf) value is the same value given for weathered rock in Table 2.5-2 of the Units 1 and 2 UFSAR (SSAR Reference 5). For dynamic loading, 20 percent of the ultimate crushing strength can be used. It should be noted that although the 958 kPa (20 ksf) allowable static bearing capacity is greater than the maximum static bearing pressure from the RB/FB basemat, the RB/FB foundation will not be founded directly on the Zone III weathered rock. If excavation during construction for this foundation reveals any weathered or fractured zones at foundation level, such zones will be over-excavated and replaced with lean concrete.

The Zone III-IV and Zone IV bedrock have design unconfined compressive strengths of 62 MPa (9 ksi (1296 ksf)) and 117 MPa (17 ksi (2448 ksf)), respectively (Table 2.5-212). The allowable static values of the bearing capacity of 3830 kPa (80 ksf) and 7660 kPa (160 ksf) for Zone III-IV and Zone IV rock, respectively, are presumptive values based on various building codes for moderately weathered to fresh foliated rock (SSAR Reference 181). For dynamic loading, 20 percent of the ultimate crushing strength can be used, i.e., 12,400 kPa (259 ksf) for Stratum III-IV, and 23,460 kPa (490 ksf) for Stratum IV. For 17 MPa (2500 psi) concrete fill, the computed allowable bearing capacity is 10,240 kPa (214 ksf) (Reference 2.5-215) for both static and dynamic loading.

b. **Soil**

For granular soils like the Zone IIB saprolite and the engineered structural fill, bearing capacity is based on Terzaghi's bearing capacity equations modified by Vesic (SSAR Reference 180). The ultimate (gross) bearing capacity of a footing, q_{ult}, supported on homogeneous soils can be estimated by (SSAR Reference 180):

$$q_{ult} = cN_c\zeta_c + \gamma' D_f N_q \zeta_q + 0.5\gamma' B N_\gamma \zeta_\gamma$$

where:

c = undrained shear strength for clay (c_u) or cohesion intercept for (c,ϕ) soil

$\gamma' D_f$ = effective overburden pressure at base of foundation

γ' = effective unit weight of soil

D_f = depth from ground surface to base of foundation

B = width of foundation

N_c, N_q, and N_γ are bearing capacity factors (defined in SSAR Reference 180), and

ζ_c, ζ_q, and ζ_γ are shape factors (defined in SSAR Reference 180)

These equations use the effective unit weight of the soil, the width and depth of the foundation, and bearing capacity and shape factors that are a function of the angle of internal friction of the soil. Consequently, each foundation has a different bearing capacity, depending on the foundation dimensions. For large foundations that are founded at large depths below grade, these equations can give very large bearing capacity values, even when a factor of safety of 3 is included for the allowable bearing value. In such situations, settlement, discussed in Section 2.5.4.10.2, normally governs.

c. **Allowable Bearing Capacity for Structures**

Table 2.5-215 gives the estimated allowable bearing capacity for the three Seismic Category I, the two Seismic Category II structures, and the radwaste building based on the materials underlying the structures shown in Figure 2.5-251 and Figure 2.5-252. Where the structure bears on soil (Zone IIB saprolite or structural fill), the theoretical allowable capacities of the soil are very large, for the reasons explained above. The design static bearing capacity given in Table 2.5-215 is generally the minimum value for any layer beneath the structure. For the CB, there may be a very limited thickness of Zone III material beneath the foundation, but this will not govern the allowable bearing capacity. The allowable static bearing capacity for this structure was conservatively chosen as 2395 kPa (50 ksf), the mean of the values for Zone III and Zone III-IV. For structures on soil, settlement estimates are needed to determine what value of bearing pressure can be realistically applied.

Table 2.5-215 also contains values of allowable bearing capacity under dynamic or transient loading conditions. For bedrock subgrade, as noted earlier, these values are equivalent to 20 percent of the ultimate crushing strength. For soils, the values represent an increase of one third over the allowable static bearing capacity values. Note that the allowable static and dynamic bearing capacity values in Table 2.5-215, for the Category I RB/FB, CB and FWSC foundations, exceed the design soil or rock applied bearing stresses given in Table 2.5-213.

The Zone IIA saprolite can be used to support relatively lightly-loaded, non-settlement sensitive structures that are not classified as Seismic Category I or II. The allowable bearing capacity value is limited to 192 kPa (4 ksf) because of settlement considerations. (The 192 kPa (4 ksf) value can be increased by one third for dynamic or transient conditions.) As noted in Section 2.5.4.10.2, settlement considerations usually dominate when this material is used for supporting foundations, and the actual allowable bearing capacity may be less than 192 kPa (4 ksf), especially for larger foundations.

d. **Groundwater Effects**

NAPS ESP COL 2.5-4

Based on the conservative assumption of the groundwater table being 0.6 m (2 ft) below grade, there can be a hydrostatic uplift force on any buried structure. All of the below-ground structures shown in Table 2.5-213 (i.e., all except the FWSC and service building) have applied foundation loads that are at least 6 ksf, and so there are no net uplift forces. However, such forces can be significant in the design of buried piping, particularly when the pipe is empty. In such a situation, the weight and strength of the backfill above the pipe is analyzed to confirm satisfactory resistance to the uplift forces. The normal factor of safety of 3 against soil failure is used in this analysis.

NAPS ESP COL 2.5-6

2.5.4.10.2 Settlement Analysis

The pseudo-elastic method of analysis was used for settlement estimates. This approach is suitable for the granular soils and bedrock at the site. The analysis is based on a stress-strain model that computes settlement of discrete layers:

$$\delta = \Sigma(\Delta p_i \times \Delta h_i)/E_i$$

where:

δ = settlement

i = 1 to n, where n is the number of soil layers

Δp_i = vertical applied pressure at center of layer i

Δh_i = thickness of layer i

E_i = elastic modulus of layer i

The stress distribution below the rectangular foundations is based on a Boussinesq-type distribution for flexible foundations (Reference 2.5-216). The computation extends to a depth where the increase in vertical stress (Δp) due to the applied load is equal to or less than 10 percent of the applied foundation pressure. The Boussinesq-type vertical pressure under a rectangular footing, σ_z, is as follows (Reference 2.5-216):

$$\sigma_z = (p/2\pi)(\tan^{-1}(lb/(zR_3)) + (lbz/R_3)(1/R_1^2 + 1/R_2^2))$$

where:

l = length of footing

b = width of footing

z = depth below footing at which pressure is computed

R1 = $(l^2 + z^2)^{0.5}$

R2 = $(b^2 + z^2)^{0.5}$

R3 = $(l^2 + b^2 + z^2)^{0.5}$

Settlement estimates were made using the preceding relationships and the soil and rock properties given in Table 2.5-212. These estimates were made for each Seismic Category I and II structure, and the radwaste building, and are presented in Table 2.5-216. The applied pressures from the foundations are shown on Table 2.5-216.

As would be anticipated, the settlement of the structures founded on Zone III-IV or Zone IV bedrock is negligible. Similarly, settlements of structures sitting on the dense to very dense structural fill or Zone IIB saprolite overlying rock are modest in light of the large applied pressures. Differential settlements within the structure are close to 50 percent of the total settlement except for the turbine building where parts of the structure are founded on bedrock and other parts are on soil. In such a case, the differential settlement within the structure can approach the total settlement value.

Note that the total and differential settlements under the RB/FB, CB and FWSC are well within the limits stated in Table 2.0-201.

2.5.4.10.3 Earth Pressures

Static and seismic lateral earth pressures are addressed for plant below-ground walls. Both active and at-rest cases are included. The earth pressure coefficients are Rankine values, assuming level backfill and a zero friction angle between the soil and the wall. Hydrostatic pressures are conservatively based on the groundwater table being 0.6 m (2 ft) below grade. A surcharge pressure of 23.9 kPa (500 psf) is used. Lateral pressures due to compaction are not included; these pressures are controlled by compacting backfill with light equipment near structures. The soil properties used in the calculation of lateral earth pressures are from Table 2.5-212.

For the active lateral earth pressure case, earthquake-induced horizontal ground accelerations are addressed by the application of $k_h \cdot g$. Vertical ground accelerations ($k_v \cdot g$) are considered negligible and were ignored (Reference 2.5-217). The peak low frequency acceleration of 0.31g was used for developing the seismic active earth pressure diagrams. Use of the peak high frequency acceleration was considered overly conservative given the low magnitude (energy) of this earthquake.

Recognizing the limitation of the Reference 2.5-217 method for design of building walls, Ostadan (Reference 2.5-218) developed a method to compute seismic soil pressure that focused on building walls rather than soil retaining walls. This method specifically considers the following: a) the movement of the walls is limited due to the presence of the floor diaphragms and the walls are considered non-yielding; b) the frequency content of the design motion is fully taken into account; and c) appropriate soil properties, in terms of soil shear wave velocity and damping, are included in the analysis. The method is flexible to allow for consideration of soil nonlinear effects where soil nonlinearity is expected to be significant. This method was used to estimate the seismic lateral at-rest pressures against the buried structure walls. The response spectrum at the bottom of the RB/FB was used in this analysis.

Figure 2.5-229 through Figure 2.5-234 show structural fill between below-ground structures, e.g., between the RB and CB in Figure 2.5-232. In this situation, the at-rest lateral pressure due to the structural fill is used to compute wall pressures. The same figure shows structural fill between the vertical excavation support wall and the below-ground RB wall. Zone IIA and IIB saprolite are on the other side of the wall and are in an active condition after excavation within the wall. In this situation, the lateral earth pressures against the vertical excavation support wall can have some influence on the earth pressure against the RB wall. Thus, active earth pressures due to the Zone IIA and IIB saprolites are included here.

Lateral earth pressure diagrams for the active and at-rest cases are given in Figure 2.5-253 and Figure 2.5-254, respectively.

Note that the lateral pressures in Figure 2.5-253 and Figure 2.5-254 are best estimate pressures with a factor of safety of 1. Appropriate safety factors need to be incorporated into the wall structural design. The factor of safety against a gravity wall or structure foundation sliding is normally taken as 1.1 when seismic pressures are included. The same factor of safety is applied against a wall overturning.

NAPS COL 2.0-29-A

2.5.4.11 Design Criteria

NAPS ESP COL 2.5-7

Applicable design criteria are covered in various sections. The criteria summarized below are geotechnical criteria and also geotechnical-related criteria that pertain to structural design.

Section 2.5.4.8 specifies that the acceptable factor of safety against liquefaction of site soils is ≥ 1.1.

Bearing capacity and settlement criteria are presented in Section 2.5.4.10. Table 2.5-215 provides allowable bearing capacity values for the Seismic Category I and II structures and the radwaste building. A minimum factor of safety of 3 is used when applying bearing capacity equations. This factor of safety is also applied against breakout failure due to uplift forces on buried piping. For soils, this factor of safety can be reduced to 2.25 when dynamic or transient loading conditions apply.

Section 2.5.4.10 also discusses factors of safety related to lateral earth pressures. The lateral pressures shown in Figure 2.5-253 and Figure 2.5-254 have a factor of safety of 1. A factor of safety of 1.1 should be used in the analyses of sliding and overturning due to these lateral loads when the seismic component is included.

Section 2.5.5.2 specifies that the minimum acceptable long-term static factor of safety against slope stability failure is 1.5. Section 2.5.5.3 specifies that the minimum acceptable long-term seismic factor of safety against slope stability failure is 1.1.

NAPS COL 2.0-29-A

2.5.4.12 Techniques to Improve Subsurface Conditions

NAPS ESP COL 2.5-8

For Unit 3, any Zone IIA saprolite beneath or within the zone of influence of Seismic Category I or II structures is removed and replaced with compacted structural fill. Improvement of the Zone IIA saprolite as described SSAR Section 2.5.4.12 is suitable for non-Seismic Category I and II structures.

Zones of weathered or fractured rock encountered immediately beneath the RB/FB basemat are removed and replaced with concrete.

Appendix 2.5.4AA MACTEC Geotechnical Data Report, Rev. 1; September 28, 2007

Volume 1: Text, Figures, Tables and Appendices A and B

Volume 2: Appendices C and D

Volume 3: Appendix E

Appendix 2.5.4AAS1 Supplement 1, Dynamic Laboratory Testing Results

Appendix 2.5.4AAS2 Supplement 2, Distribution Coefficients (Kd) Laboratory Test Results

2.5.5 Stability of Slopes

NAPS COL 2.0-30-A

The information needed to address DCD COL Item 2.0-30-A is included in the following sections.

SSAR Section 2.5.5 is incorporated by reference with the following variances and/or supplements.

NAPS ESP VAR 2.5-1

SSAR Section 2.5.5 addressed the stability of slopes at the North Anna ESP site. However, the information presented in this FSAR section replaces the analyses presented in SSAR Section 2.5.5 because the slopes being considered have changed, and, for the seismic slope stability analysis, the peak ground acceleration being applied is different. The method of analysis remains essentially the same. In summary, the slopes considered herein are lower, less steep, and have a smaller applied seismic acceleration than the slopes analyzed in SSAR Section 2.5.5. As a result, the slopes addressed in this section have a higher computed factor of safety against failure, and are stable under both long-term static and short-term seismic conditions.

This section presents information on the stability of permanent slopes at the Unit 3 site. The information was developed from a review of reports prepared for the existing units and the originally planned Units 3 and 4, geotechnical literature, the ESP subsurface investigation, and the Unit 3 subsurface investigation. The review included the site-specific reports from the UFSAR (SSAR Reference 5), and reports prepared by Dames and Moore regarding the design and construction of the existing units (SSAR Reference 7) and the originally planned Units 3 and 4 (SSAR Reference 8).

a. **Description of Slopes**

The grading plan for Unit 3 is shown in Figure 2.5-255. The design plant grade for the power block area is at Elevation 88.4 m (290 ft) with elevations around the perimeter of this area ranging from about Elevation 88.1 m (289 ft) to 86.6 m (284 ft) to allow for adequate surface drainage. To the northeast of the power block area, going towards the existing Units 1 and 2, ground surface elevation reduces at a 2 percent slope down to the yard grade of Units 1 and 2 at Elevation 82.3 m (270 ft). (Coordinates and directions in this section are with reference to true north.) To attain these ground elevations, there is cut in the power block area, reaching as much as 12.2 m (40 ft) to the south of the reactor building. However, as existing grade falls off towards the northeast of the power block area, there is as much as 6.1 m (20 ft) of fill needed around portions of the northeast end of the turbine building. As much as 9.1 m (30 ft) of fill is provided to bring grade up to the planned ground surface in the area of the originally planned Units 3 and 4, where ground level is presently at around Elevation 76.2 m (250 ft).

As shown in Figure 2.5-255, there are no slopes that contribute to the support of any of the Unit 3 seismic Category I structures or any of the other major powerblock structures. The only slopes that could impact Unit 3 are cut slopes that surround and ascend from the southern edges of the plant. As discussed in Section 2.5.5b, material from sloughing or collapse of certain of these slopes could impact certain facilities. These new slopes are cut at a 3-horizontal to 1-vertical (3h:1v) slope into the existing natural ground surrounding the plant, with a 4.6 m (15 ft) wide bench constructed at about 6.1 m (20 ft) height from the bottom of the slope. These slopes reach a maximum height of 14.6 m (48 ft) (from Elevation 87.2 m (286 ft) up to Elevation 101.8 m (334 ft)) southwest of the plant, to the northwest of the administration building. The height of the slope reduces to the southeast of the plant. Southeast of the FWSC, the height is about 10.7 m (35 ft) (from Elevation 87.5 m (287 ft) up to Elevation 98.1 m (322 ft)). This is identified as Slope A-A in Figure 2.5-255.

The new cut slope to the southeast of the FWSC merges into an existing slope (see Slope ES in Figure 2.5-255) that runs in a northeasterly direction, to the south of the originally planned Units 3 and 4 and existing Units 1 and 2. Based on previous topographic maps, this slope was described in the SSAR as a 2h:1v slope, 16.8 m (55 ft) high. A new topographic survey performed for Unit 3 shows that the slope is actually about 2.4h:1v with a maximum height of 15.8 m (52 ft) (from Elevation 82.6 m (271 ft) to Elevation 98.5 m (323 ft)). Based on the final grade for Unit 3, the maximum height of this existing slope within the vicinity of any new structures is south of the service water cooling tower, where the height is about 13.1 m (43 ft) (from Elevation 85.3 m (280 ft) to Elevation 98.5 m (323 ft)).

The maximum depth of the storm water basin to the northeast of the main plant area is 6.7 m (22 ft) (from Elevation 86.0 m (282 ft) down to Elevation 79.2 m (260 ft)). This basin is cut at a 3h:1v slope.

SSAR Section 2.5.5 refers to slopes resulting from the nonsafety-related deepened intake channel. In fact, the intake channel for Unit 3 will not be deepened, and thus there will be no new slopes associated with the intake channel.

As shown in Figures 2.5-229 through 2.5-234, temporary excavation for Unit 3 construction will be performed using tied-back vertical walls.

b. **Impact of Slope Instability**

Instability of the storm water basin sides does not impact the safety of the plant, nor any of the other plant structures, and so such slopes are not addressed further here. Failure of any temporary slope or excavation created for construction of the plant cannot adversely affect the safety of the nuclear power plant facilities, and likewise this is not addressed further here.

The existing 2.4h:1v slope (Slope ES) was excavated during construction of the Units 1 and 2, and is almost entirely in cut material. The top of this slope is about 61 m (200 ft) from the top of the existing service water reservoir (SWR) embankment, and thus any potential instability of the slope will have no impact on the stability of the SWR embankment. However, material from sloughing or collapse of these slopes could impact the new diesel tanks and/or service water cooling tower. Slope ES is a representative section along the approximately 215 m (700 ft) length of the existing slope.

Instability of the new 3h:1v slope to the southeast of the FWSC (Slope A-A) does not impact the foundation stability of this Seismic Category I facility since the facility is founded on stable compacted crushed rock fill. However, material from sloughing or collapse of this slope could impact the facility, because the base of this new 10.7 m (35 ft) high slope is about 16.8 m (55 ft) from the FWSC. As can be seen from Figure 2.5-255, the new slopes extend to the south of Slope A-A and then west to northwest past the administration building, which is built into the slope. Although these slopes are somewhat higher than Slope A-A, they are much farther away from the Seismic Category I structures, and sloughing or collapse of these slopes would not impact any of the Seismic Category I structures. Thus, Slope A-A is considered the critical slope in the area.

The stability of the existing slope closest to the new service water cooling tower (Slope ES), and the stability of the new slope closest to the FWSC (Slope A-A) are addressed in the following subsections.

2.5.5.1 Slope characteristics

2.5.5.1.1 Existing Slope Characteristics

The location and direction of the existing 13.1 m (43 ft) high, 2.4h:1v slope to the north of the Units 1 and 2 SWR (Slope ES) is shown in plan view in Figure 2.5-255; the location is also shown in the photograph in SSAR Figure 2.5-66. The photograph in SSAR Figure 2.5-67 shows the existing slope clearly, descending from the SWR to close to the excavation for the originally planned Unit 3 and 4 containment buildings. The structure behind the slope on the SWR embankment is the Units 1 and 2 valve house, which was initially designed to be the originally planned Units 3 and 4 pump house. An approximate cross-section through the existing slope is shown in Figure 2.5-256.

As shown in Figures 2.5-255 and 2.5-256, a boring (B-18) was drilled close to the toe of the slope. This boring was made for the Units 1 and 2 investigation. During the Unit 3 subsurface investigation, cone penetrometer test (CPT) C-915 was performed near to the top of the slope. Also during the Unit 3 investigation, boring B-947 was drilled to the west of C-915, but at a similar elevation within the same original terrain as C-915. CPT C-916 and observation well OW-947 were located adjacent to B-947. The locations of boring B-18 and CPT C-915 are included in Figure 2.5-256, along with the ground water level measured in OW-947. The boring and CPT logs are presented in Section 2.5.5.3.

2.5.5.1.2 New Slope Characteristics

NAPS ESP COL 2.5-11 The location of the new 10.7 m (35 ft) high, 3h:1v slope to the southeast of the FWSC (Slope A-A) is shown in plan view in Figure 2.5-255. An approximate cross-section through the new slope is shown in Figure 2.5-257. As shown in Figure 2.5-255, boring B-947 was drilled relatively close to the final location of the top of the slope during the Unit 3 subsurface investigation. CPT C-916 and observation well OW-947 were located adjacent to B-947. The boring and CPT logs are presented in Section 2.5.5.3. The stability analysis performed for Slope A-A (Section 2.5.5.2.4) conservatively neglected the 4.6 m (15 ft) wide bench in the slope. For consistency, this bench is not shown in Figure 2.5-257.

NAPS COL 2.0-30-A

2.5.5.1.3 Slope Subsurface Conditions

The site soils and bedrock are described in detail in Section 2.5.4.2.2. As can be seen from Figures 2.5-256 and 2.5-257, the materials in the existing and new slopes, respectively, consist mostly of Zone IIA saprolites. Saprolites are a further stage of weathering beyond weathered rock. They have been derived by in-place disintegration and decomposition and have not been transported. Saprolites are classified as soils but still contain the relict structure of the parent rock, and they also typically still contain some core stone of the parent rock. The North Anna saprolites in many instances maintain the foliation (banded texture) characteristics of the parent rock. The majority of the saprolites in the Unit 3 area are classified as silty sands, although there are also sands, clayey sands, sandy silts, clayey silts and clays, depending very much on their degree of weathering. The fabric is strongly anisotropic. The texture shows angular geometrically interlocking grains with a lack of void network, very unlike the well-pronounced voids found in marine or alluvial sands and silts. The Zone IIA saprolites comprise a large majority of the saprolitic materials onsite. Most of the saprolites obtained from the borings in the slope area are medium dense to dense silty sands. The underlying Zone IIB saprolites are generally very dense silty sands.

Boring B-18 in Figure 2.5-256 indicates top of bedrock levels rising significantly towards the toe of the existing slope, with top of weathered rock close to the slope surface at the B-18 location at around Elevation 88.4 m (290 ft). This is consistent with the top of bedrock levels shown in Figure 2.5-2, from SSAR Reference 5. For the new slope shown in Figure 2.5-257, the top of weathered rock is closer to Elevation 76.2 m (250 ft). The bedrock at North Anna ranges from moderately to severely weathered (Zone III) as encountered in B-18, to fresh to slightly weathered (Zone IV). The bedrock throughout the North Anna site is classified as a gneiss, which is a metamorphic rock that exhibits foliation in which light and dark bands alternate. It is composed of feldspar, quartz, and one or more other minerals such as mica and hornblende.

The engineering properties of the site soils and bedrock are described in Section 2.5.4.2.5 and are tabulated in Table 2.5-212. These properties are based on extensive field and laboratory testing described in Section 2.5.4.2.

The liquefaction characteristics of all of the Zone IIA saprolites are thoroughly examined in Section 2.5.4.8. That section concludes that the results of the liquefaction analysis indicate that only a very limited amount of the Zone IIA saprolitic soil has a potential for liquefaction based on the Unit 3 seismic parameters. The liquefaction analysis did not take into account the beneficial effects of age, structure, fabric, and mineralogy of the saprolitic soils.

Details of the soils encountered in the new and existing slopes are outlined in the following paragraphs.

Boring B-947, close to the top of the new slope, indicates a predominantly silty sand profile, alternating with layers of silt in the top 4.6 m (15 ft) (boring and CPT logs are presented in Section 2.5.5.3). Grain size analyses performed on 10 samples ranging in depth from 1.5 m (5 ft) to 13.1 m (43 ft) (see Section 2.5.5.3) showed fines contents varying from about 14 to 70 percent, with a median of about 29 percent. The bottom 3.0 m (10 ft) of soil has an adjusted SPT N-value of over 50 blows/0.3 m (1 ft), which is characteristic of Zone IIB saprolite. The overlying soils are Zone IIA saprolites. Interpretation of CPT C-916 (performed adjacent to B-947) based on friction ratio, indicates mainly silty clays and clays, and thus, for these saprolites, this interpretation indicates a less granular composition than shown in the grain size tests.

For stability analyses of the new slope presented in Section 2.5.5.2, based on the results of B-947 and C-916, the new slope has the properties of Zone IIA silty sand saprolite given in Table 2.5-212 down to about 10.7 m (35 ft) below existing ground level. The bottom 3.0 m (10 ft) of saprolite above weathered rock has the Zone IIB saprolite properties given in Table 2.5-212.

The log of CPT-915 (located close to the top of the existing slope) is very similar to the log of CPT-916 in the top 9.1 m (30 ft). CPT-915 continues in a similar pattern below 9.1 m (30 ft) down to 15.2 m (50 ft) where it shows significantly increased tip resistance. Below 9.1 m (30 ft), C-916 shows higher tip resistance values than C-915 down to 15.2 m (50 ft) depth.

For the stability analysis of the existing slope presented in Section 2.5.5.2, based on the results of C-915 in comparison with C-916, the existing slope has the properties of Zone IIA silty sand saprolite given in Table 2.5-212 down to about 16.8 m (55 ft) below existing ground level. The thickness of Zone IIB saprolite below the Zone IIA material becomes less towards the toe of the slope and this layer eventually pinches out as the top of weathered rock rises, as postulated in Figure 2.5-256. The Zone IIB saprolite and the weathered rock have the properties given in Table 2.5-212.

2.5.5.1.4 Slope Phreatic Surface

The phreatic surfaces shown in Figure 2.5-256 (existing slope) and Figure 2.5-257 (new slope) have been developed from the water table levels measured in OW-947 and derived in Section 2.4.12. The depth of this phreatic surface precludes any potential for liquefaction of the near-surface soils in the slope.

2.5.5.2 Design Criteria and Analyses

2.5.5.2.1 Required Factor of Safety

Factors of safety for the required stability of slopes are provided in DCD Table 2.0-1. Minimum factors of safety for static (non-seismic) loading and for dynamic (seismic) loading are 1.5 and 1.1, respectively.

2.5.5.2.2 Stability of Existing Slope

The photograph in SSAR Figure 2.5-67 of the existing 2.4h:1v slope to the north of the SWR was taken over 20 years ago. The condition of the slope is essentially the same today. It was thoroughly inspected during the ESP site investigation. The slope shows no signs of distress.

2.5.5.2.3 Analysis of Existing Slope

The static and dynamic stability of the existing slope to the north of the SWR was analyzed using the computer program SLOPE/W (Reference 2.5-219).

a. **Long-Term Static Analysis**

The SLOPE/W Program used the Bishop method of slices (SSAR Reference 185) for analysis of the long-term static condition. As noted in Section 2.5.5.1.3, the analysis assumed the saprolite was predominantly coarse grained. The effective strength parameters given in Table 2.5-212 are an angle of internal friction ϕ' = 33 degrees and effective cohesion c' = 6.0kPa (0.125 ksf) for the Zone IIA saprolite and ϕ' = 40 degrees and effective cohesion c' = 0 kPa (0 ksf) for the Zone IIB saprolite. The underlying weathered rock used c = 3350 kPa (70 ksf), approximately half of the value for unconfined compressive strength given in Table 2.5-212.

The input to the analysis and the results are shown in Figure 2.5-258. The computed factor of safety is 2.09. This value is above the minimum 1.5 factor of safety required.

b. **Seismic Slope Stability Analysis**

NAPS ESP COL 2.5-10

The pseudo-static approach is used as a first approximation for the seismic analysis of slopes. In this approach, the horizontal and vertical seismic forces are assumed to act on the slope in a static manner, that is, as a constant static force. This is an obviously conservative approach, since the actual seismic event occurs for only a short period of time, and during that time, the forces alternate their direction at a relatively high frequency. Also, the pseudo-static analysis tends to be run using the peak seismic acceleration; the mean acceleration during the design seismic event can be significantly less than the peak value. A pseudo-static analysis using peak acceleration values can be a useful tool in a limit analysis where the peak acceleration is relatively low. In such analyses, the computed factor of safety may well exceed the minimum of 1.1, thus requiring no further analysis. However, where the peak seismic acceleration values are high, the pseudo-static analysis produces unreasonably low safety factor values.

The pseudo-static analysis was run on the existing 13.1 m (43 ft) high slope (Slope ES) using SLOPE/W with the Bishop method of slices. For the low frequency earthquake, the peak horizontal acceleration used was about 0.23g. This is the average peak acceleration in the top 13.1 m (43 ft) of soil shown in Table 2.5-217. (The peak horizontal acceleration is 0.31g at the ground surface.) The vertical acceleration used was about 0.115g. The computed factor of safety was 1.29, more than the minimum 1.1 required. For the high frequency earthquake, the equivalent peak horizontal acceleration used was 0.50g with a vertical acceleration of 0.25g. The computed factor of safety was about 0.90, less than the minimum 1.1 required. The input to the analysis, and the results, are shown in Figure 2.5-259 for the low frequency earthquake and Figure 2.5-260 for the high frequency earthquake.

Seed (SSAR Reference 186), in the 19th Rankine Lecture, addressed the over-conservatism intrinsic in the pseudo-static analysis. He looked at the more rational approach proposed by Newmark (SSAR Reference 187), where the effective acceleration time-history is integrated to determine velocities and displacements of the slope. He also examined dams in California that had been subjected to seismic forces, including several dams that survived the 1906 San Francisco earthquake. Based on his studies, he concluded that for embankments that consist of materials that do not tend to build up large pore pressures or lose significant percentages of their shear strength during seismic shaking, seismic coefficients of only 0.15g are adequate to ensure acceptable embankment performance for earthquakes up to Magnitude M = 8.25 with peak ground accelerations of 0.75g. For earthquakes in the range of M = 6.5, Seed recommends a horizontal seismic coefficient of only 0.1g with a vertical seismic coefficient of zero. Note that it is the magnitude of the earthquake that determines the acceleration to be used here; magnitude is not part of the input to the pseudo-static analysis.

The liquefaction analysis of the Zone IIA saprolite indicated that only a very limited amount of the material has a potential for liquefaction. Also, because of its age, fabric and interlocking angular grain structure, this material does not lose a significant proportion of its shear strength during shaking. Thus, for the low frequency earthquake, with a design Magnitude M = 7.2, the pseudo-static analysis should be limited to a horizontal acceleration of 0.15g. A pseudo-static design using an inertia force of 0.1g is adequate for the high frequency earthquake.

The pseudo-static analysis was run again using SLOPE/W. This time the horizontal accelerations used were 0.1g and 0.15g, with zero vertical acceleration. The computed factors of safety were 1.63 and 1.47, respectively, greater than the minimum 1.1 required. The input to the analysis, and the results, for the 0.1g and 0.15g cases are shown in Figure 2.5-261 and 2.5-262, respectively.

Other researchers have also recommended substantially reducing the peak acceleration when applying the pseudo-static analysis. Kramer (SSAR Reference 188) recommends using an acceleration of 50 percent of the peak acceleration. For the low frequency earthquake, where the average peak acceleration in the top 13.1 m (43 ft) is about 0.23g, the horizontal input using Kramer’s recommendations was about 0.115g and the vertical input was about 0.058g. This results in a factor of safety of

1.59. Using the average peak acceleration for the high frequency earthquake in the top 13.1 m (43 ft) of 0.50g, the horizontal input using Kramer's recommendation was 0.25g and the vertical input was 0.125g. This level of input provides a factor of safety against slope failure of 1.24. Thus the low and high frequency inputs give factors of safety above the minimum 1.1 required. The input to the analysis, and the results, for the low frequency and high frequency cases are shown in Figure 2.5-263 and 2.5-264, respectively.

In the preceding analyses (both long-term static, and seismic), the only case that gave a factor of safety lower than the required minimum was the pseudo-static analysis using the high frequency peak acceleration. As noted above, the pseudo-static analysis does not take into account the frequency of the motion nor the magnitude of the earthquake. For high frequency, low magnitude earthquakes, (as is the case at North Anna) the pseudo-static analysis is particularly conservative. Thus, it is concluded that the existing 2.4h:1v slope to the north of the SWR remains stable under long-term static and design seismic conditions.

NAPS COL 2.0-30-A

2.5.5.2.4 Analysis of New Slope

The static and dynamic stability of the new 10.7 m (35 ft) high 3h:1v slope (Slope A-A) to the southeast of the FWSC was analyzed using the computer program SLOPE/W (Reference 2.5-219).

a. **Long-Term Static Analysis**

The SLOPE/W Program used the Bishop method of slices (SSAR Reference 185) for analysis of the long-term static condition. As noted in Section 2.5.5.1.3, the properties assumed for the Zone IIA and Zone IIB saprolite were the same as those for the existing slope that was analyzed.

The input to the analysis and the results are shown in Figure 2.5-265. The computed factor of safety is 2.23. This value is above the minimum 1.5 factor of safety required.

b. **Seismic Slope Stability Analysis**

NAPS ESP COL 2.5-10

The pseudo-static analysis was run on the new 10.7 m (35 ft) high slope using SLOPE/W with the Bishop method of slices. For the low frequency earthquake, the average peak horizontal acceleration in the top 10.7 m (35 ft) used was about 0.23g with a vertical acceleration of about 0.115g. The computed factor of safety was 1.30, greater than the minimum 1.1

required. For the high frequency earthquake, the peak horizontal acceleration used was about 0.50g. This is the average peak acceleration in the top 10.7 m (35 ft) of soil shown in Table 2.5-217. (The maximum horizontal acceleration is 0.55g at the ground surface.) The vertical acceleration used was about 0.25g. The computed factor of safety was 0.90, less than the minimum 1.1 required. The input to the analysis, and the results, for the low frequency and high frequency cases are shown in Figure 2.5-266 and 2.5-267, respectively.

The pseudo-static analysis was run again using SLOPE/W and Seed's (SSAR Reference 186) approach described in Section 2.5.5.2.3. Again the horizontal accelerations used were 0.1g and 0.15g for the high and low frequency cases, respectively, with zero vertical acceleration. The computed factors of safety were 1.64 and 1.44, respectively, greater than the minimum 1.1. The input to the analysis, and the results, for the 0.1g and 0.15g cases are shown in Figure 2.5-268 and 2.5-269, respectively.

The pseudo-static analysis was then run using SLOPE/W and Kramer's (SSAR Reference 188) recommendations described in Section 2.5.5.2.3. For the low frequency earthquake, where the average peak acceleration in the top 10.7 m (35 ft) is about 0.23g, the horizontal input using Kramer's recommendations was about 0.115g and the vertical input was about 0.06g. Using the average peak acceleration for the high frequency earthquake in the top 10.7 m (35 ft) of about 0.50g, the horizontal input using Kramer's recommendation was 0.25g and the vertical input was 0.125g. These levels of input provide a factor of safety against slope failure of 1.63 and 1.25 for the low and high frequency cases, respectively, greater than the minimum 1.1 required. The input to the analysis, and the results, for the low frequency and high frequency cases are shown in Figure 2.5-270 and 2.5-271, respectively.

The results of the stability analyses for the new slope are almost identical to those for the existing slope, and the conclusion about stability is the same, i.e., the new 3h:1v slope to the southeast of the FWSC remains stable under long-term static and design seismic conditions.

NAPS COL 2.0-30-A

2.5.5.3 Logs of Borings

2.5.5.3.1 Boring Logs

As noted in Section 2.5.5.1, boring B-18 was drilled close to the toe of the existing 2.4h:1v slope to the north of the SWR and boring B-947 was drilled near the top of the proposed new 3h:1v slope southeast of the FWSC. The logs of borings B-18 and B-947 are reproduced in Figure 2.5-272 and 2.5-273, respectively.

2.5.5.3.2 CPT Logs

As noted in Section 2.5.5.1, CPT C-915 was drilled close to the top of the existing 2.4h:1v slope to the north of the SWR and CPT C-916 was drilled adjacent to B-947 near the top of the new 3h:1v slope southeast of the FWSC. The logs of CPTs C-915 and C-916 are reproduced in Figure 2.5-274 and 2.5-275, respectively.

2.5.5.3.3 Observation Wells

As noted in Section 2.5.5.1, observation well OW-947 was installed adjacent to boring B-947 near the top of the new 3h:1v slope southeast of the FWSC. The log of OW-947 is reproduced in Figure 2.5-276. Water levels measured in this well over a 12-month period are shown in Table 2.5-218.

2.5.5.3.4 Laboratory Test Results

The grain size tests results for the saprolites in boring B-947 and noted in Section 2.5.5.1 are provided in Table 2.5-219. Details of these test results are provided in Appendix 2.5.4AA.

2.5.5.4 Compacted Fill

The existing 2.4h:1v slope described and analyzed in the previous sections is a cut slope and does not contain fill materials in any significant quantity.

As shown in Figure 2.5-257, the grading plan results in the top approximately 2.1 m (7 ft) of the new 3h:1v slope southeast of the FWSC being new fill. This is not structural fill since it is used only for site grading and consists of re-compacted saprolitic soils obtained from plant excavations. These are described in Section 2.5.4.5. For slope stability analysis, this fill has been given the same properties as the in-situ Zone IIA saprolite.

2.5.5.5 Conclusions

Existing slopes and embankments that are not impacted by Unit 3 (such as the SWR embankments) do not require analysis for Unit 3 and are not addressed here. New slopes, such as in the storm water basin that will not impact the safety of the plant or any other structure if they fail also do not require analysis and are not addressed here. Failure of any temporary slope or excavation created for construction of Unit 3 cannot adversely affect the safety of Unit 3, consequently, this is not addressed further here.

The only existing slope which, by its failure, could adversely affect the safety of Unit 3, because of its proximity, is the 13.1 m (43 ft) high, 2.4h:1v slope that descends from north of the SWR down to south of the existing excavation made for the originally planned Units 3 and 4. The slope is made almost entirely in cut material. The stability of this existing slope was analyzed using the computer program SLOPE/W. The only case that gave a factor of safety lower than the required minimum was the pseudo-static analysis using the high frequency peak acceleration. This analysis does not take into account the frequency of the motion or the magnitude of the earthquake. For high frequency, low magnitude earthquakes, (as is the case at North Anna) the pseudo-static analysis is particularly conservative. Thus, it is concluded that this slope remains stable under long-term static and design seismic conditions.

The results of the stability analyses for the new 3h:1v slope to the southeast of the FWSC are almost identical to those for the existing slope described above, and the conclusion about stability is the same, i.e., the new slope remains stable under long-term static and design seismic conditions.

2.5.6 Embankments and Dams

SSAR Section 2.5.6 is incorporated by reference with the following supplement.

This SSAR section is supplemented as follows with a new paragraph on Unit 3 embankments and dams.

NAPS COL 2.0-30-A

Because Lake Anna is only used as a source of makeup water for Unit 3, the North Anna Dam, which is designed and constructed to meet requirements for a seismic Category I structure in support of the existing Units 1 and 2, was not re-analyzed as part of this FSAR. Construction of Unit 3 does not adversely affect the slopes of the SWR for Units 1 and 2. There is an existing slope to the north of the SWR and a new slope to the southeast of the FWSC. These slopes are described and analyzed in Section 2.5.5.

Section 2.5 References

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2.5-202 U.S. NRC. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 2.5.4, Revision 3, March 2007.

2.5-203 U.S. NRC. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," June 2007.

2.5-204 ASME NQA-1 Subpart 2.20. "Quality Assurance Program Requirements for Nuclear Facilities," 1994.

2.5-205 ASTM International. ASTM D 4220-95, Standard Practice for Preserving and Transporting Soil Samples, West Conshohocken, PA, 1995.

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Table 2.5-201 Selected Horizontal Response Spectrum Amplitudes, V/H Spectral Ratios from SSAR Reference 171, and Resulting Vertical Response Spectrum Amplitudes for the Control Point, Zone III-IV, Top of Competent Rock (Elevation 83.2 m (273 ft))

Frequency (Hz)	Horizontal - SA (g)	V/H Spectral Ratio	Vertical - SA (g)
100.00	0.448	1.00	0.448
50.00	0.969	1.12[a]	1.085
30.00	1.206	0.94[a]	1.134
25.00	1.193	0.88	1.050
20.00	1.163	0.83[a]	0.965
10.00	0.877	0.75	0.658
8.00	0.687	0.75	0.515
6.00	0.468	0.75	0.351
5.00	0.367	0.75	0.275
4.00	0.283	0.75	0.212
3.00	0.214	0.75	0.161
2.50	0.18	0.75	0.135
2.00	0.143	0.75	0.107
1.00	0.0676	0.75	0.0507
0.80	0.0578	0.75	0.0434
0.60	0.0492	0.75	0.0369
0.50	0.0432	0.75	0.0324
0.40	0.0344	0.75	0.0258
0.30	0.0234	0.75	0.0176
0.20	0.0131	0.75	0.00983
0.10	0.00386	0.75	0.00290

a. V/H ratios calculated by log-log interpretation

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Table 2.5-202 Selected Horizontal Response Spectrum Amplitudes, V/H Spectral Ratios from SSAR Reference 171, and Resulting Vertical Response Spectrum Amplitudes for the Base of the CB Foundation (Elevation 73.5 m (241 ft))

Frequency (Hz)	Horizontal - SA (g)	V/H Spectral Ratio	Vertical - SA (g)
100.00	0.433	1.00	0.433
50.00	0.962	1.12[a]	1.077
30.00	1.158	0.94[a]	1.089
25.00	1.151	0.88	1.013
20.00	1.135	0.83[a]	0.942
10.00	0.872	0.75	0.654
8.00	0.685	0.75	0.514
6.00	0.468	0.75	0.351
5.00	0.368	0.75	0.276
4.00	0.283	0.75	0.212
3.00	0.214	0.75	0.161
2.50	0.18	0.75	0.135
2.00	0.143	0.75	0.107
1.00	0.0676	0.75	0.0507
0.80	0.0578	0.75	0.0434
0.60	0.0492	0.75	0.0369
0.50	0.0432	0.75	0.0324
0.40	0.0344	0.75	0.0258
0.30	0.0234	0.75	0.0176
0.20	0.0131	0.75	0.00983
0.10	0.00385	0.75	0.00289

a. V/H ratios calculated by log-log interpretation

NAPS COL 2.0-27-A

Table 2.5-203 Selected Horizontal Response Spectrum Amplitudes, V/H Spectral Ratios from SSAR Reference 171, and Resulting Vertical Response Spectrum Amplitudes for the Base of the RB/FB Foundation (Elevation 68.3 m (224 ft))

Frequency (Hz)	Horizontal - SA (g)	V/H Spectral Ratio	Vertical - SA (g)
100.00	0.434	1.00	0.434
50.00	0.979	1.12[a]	1.096
30.00	1.174	0.94[a]	1.104
25.00	1.155	0.88	1.016
20.00	1.128	0.83[a]	0.936
10.00	0.868	0.75	0.651
8.00	0.684	0.75	0.513
6.00	0.468	0.75	0.351
5.00	0.368	0.75	0.276
4.00	0.283	0.75	0.212
3.00	0.214	0.75	0.161
2.50	0.18	0.75	0.135
2.00	0.143	0.75	0.107
1.00	0.0676	0.75	0.0507
0.80	0.0579	0.75	0.0434
0.60	0.0492	0.75	0.0369
0.50	0.0432	0.75	0.0324
0.40	0.0344	0.75	0.0258
0.30	0.0234	0.75	0.0176
0.20	0.0131	0.75	0.00983
0.10	0.00386	0.75	0.00290

a. V/H ratios calculated by log-log interpretation

NAPS COL 2.0-27-A

Table 2.5-204 Selected Horizontal Response Spectrum Amplitudes, V/H Spectral Ratios from SSAR Reference 171, and Resulting Vertical Response Spectrum Amplitudes a the Base of the FWSC Foundation (Elevation 86.0 m (282 ft))

Frequency (Hz)	Horizontal - SA (g)	V/H Spectral Ratio	Vertical - SA (g)
100.00	0.427	1.00	0.427
50.00	0.545	1.12[a]	0.610
30.00	1.887	0.94[a]	0.834
25.00	1.027	0.88	0.904
20.00	1.155	0.83[a]	0.958
10.00	1.910	0.75	0.683
8.00	0.812	0.75	0.609
6.00	0.780	0.75	0.585
5.00	0.783	0.75	0.588
4.00	0.746	0.75	0.560
3.00	0.565	0.75	0.424
2.50	0.409	0.75	0.307
2.00	0.249	0.75	0.187
1.00	0.0744	0.75	0.0558
0.80	0.0626	0.75	0.0469
0.60	0.0511	0.75	0.0383
0.50	0.0436	0.75	0.0327
0.40	0.0345	0.75	0.0259
0.30	0.0242	0.75	0.0182
0.20	0.0131	0.75	0.00984
0.10	0.00419	0.75	0.00314

a. V/H ratios calculated by log-log interpretation

NAPS COL 2.0-29A

Table 2.5-205 Borehole Information

Boring Number	Coordinates (ft) Northing	Coordinates (ft) Easting	Ground Surface Elevation (ft)	Penetration Depth (ft)
B-901	3,909,777.72	11,685,928.59	309.42	300.0
B-902	3,909,874.19	11,685,884.28	302.20	201.7
B-903	3,909,812.10	11,686,028.80	301.59	151.0
B-904	3,909,692.47	11,685,970.43	316.75	151.7
B-905	3,909,732.86	11,685,821.97	306.75	150.4
B-906	3,909,670.03	11,685,795.34	311.72	150.5
B-907	3,909,607.90	11,685,938.35	322.71	200.5
B-908	3,909,716.65	11,686,060.89	307.71	151.4
B-909	3,909,695.46	11,686,107.40	304.90	201.9
B-910	3,909,667.63	11,685,883.11	316.54	148.4
B-911	3,909,919.91	11,685,992.68	299.79	101.0
B-911A	3,909,916.04	11,686,000.53	299.91	21.7
B-912	3,910,021.70	11,686,051.36	275.10	151.8
B-913	3,910,148.50	11,686,114.71	273.37	100.9
B-914	3,909,939.55	11,685,922.35	297.45	200.5
B-915	3,909,877.48	11,686,088.55	301.79	112.8
B-916	3,910,049.54	11,686,008.70	276.24	100.3
B-917	3,910,160.68	11,686,029.45	274.85	150.8
B-918	3,910,115.28	11,686,194.05	272.13	150.1
B-919	3,909,575.39	11,685,764.67	317.79	76.2
B-920	3,909,545.07	11,685,980.20	327.17	150.7
B-921	3,909,680.19	11,686,162.71	307.96	73.9
B-921A	3,909,686.89	11,686,161.68	307.39	40.4
B-922	3,909,943.65	11,686,232.99	271.30	26.0
B-922A	3,909,949.30	11,686,244.02	271.33	76.5
B-923	3,910,076.97	11,686,309.48	272.00	75.4
B-924	3,909,969.53	11,686,475.40	271.52	75.6
B-925	3,910,036.67	11,686,576.27	270.01	75.8

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Table 2.5-205 Borehole Information

Boring Number	Coordinates (ft)		Ground Surface Elevation (ft)	Penetration Depth (ft)
	Northing	Easting		
B-926	3,910,043.20	11,685,709.26	289.03	155.5
B-927	3,909,966.07	11,685,878.59	292.51	100.4
B-928	3,910,222.75	11,686,159.07	272.17	75.2
B-928A	3,910,220.39	11,686,165.35	271.82	37.5
B-929	3,909,214.44	11,685,654.82	329.02	74.0
B-929A	3,909,214.15	11,685,665.51	329.03	52.5
B-930	3,909,275.95	11,685,842.87	326.12	123.6
B-931	3,910,152.94	11,685,921.54	278.52	74.0
B-932	3,910,444.31	11,686,415.70	249.88	35.1
B-933	3,909,827.41	11,685,790.97	296.48	100.3
B-933A	3,909,826.28	11,685,802.01	296.58	27.5
B-934	3,909,860.37	11,685,686.09	294.80	101.6
B-936	3,910,745.87	11,685,929.15	286.56	100.7
B-937	3,910,688.52	11,686,672.12	270.25	55.3
B-939	3,911,317.60	11,686,605.91	254.03	76.1
B-940	3,910,266.77	11,688,901.02	268.32	76.1
B-941	3,910,403.63	11,688,912.87	267.19	75.8
B-942	3,909,614.69	11,684,326.45	291.85	100.8
B-943	3,909,355.39	11,683,892.47	300.40	101.9
B-944	3,908,772.38	11,684,127.62	334.69	86.4
B-945	3,910,135.55	11,683,779.79	281.51	100.6
B-946	3,908,787.24	11,683,810.59	333.36	100.7
B-947	3,909,574.53	11,686,367.21	312.48	88.8
B-948	3,909,619.26	11,685,565.69	310.41	100.6
B-949	3,909,018.09	11,685,157.27	334.82	106.4
B-950	3,910,835.82	11,686,282.11	282.50	100.8
B-951	3,910,548.26	11,686,821.80	249.93	101.0

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Table 2.5-206 Observation Well Information

Well Number	Coordinates (ft)		Surface Elev. (ft)	Depth (ft)	Elev. of Top of Screen (ft)	Screen Length (ft)
	Northing	Easting				
OW-901	3,909,772	11,685,917	309.6	108.0	214.6	10
OW-945	3,910,136	11,683,793	281.6	54.5	240.1	10
OW-946	3,908,788	11,683,823	334.0	43.4	303.6	10
OW-947	3,909,580	11,686,372	313.3	58.0	268.3	10
OW-949	3,909,025	11,685,153	335.7	105.0	243.2	0
OW-950	3,910,842	11,686,285	283.0	92.0	203.0	10
OW-951	3,910,521	11,686,786	249.7	67.0	194.6	10

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Table 2.5-207 Information on the CPTs Performed

CPT Number	Coordinates (ft)		Ground Surface Elevation (ft)	Depth (ft)
	Northing	Easting		
C-901	3,909,627.77	11,686,012.67	318.56	20.0
C-902	3,909,552.59	11,685,842.21	323.66	29.0
C-903	3,909,719.02	11,685,775.66	306.84	29.0
C-904	3,910,026.29	11,685,793.52	283.92	35.5
C-905	3,910,137.61	11,685,857.21	279.29	45.6
C-906	3,910,013.77	11,686,269.94	270.75	2.6
C-907	3,910,174.67	11,686,277.14	271.66	13.1
C-908	3,910,326.76	11,686,187.39	271.91	28.1
C-909	3,909,346.74	11,685,717.77	330.26	60.0
C-910	3,909,154.43	11,685,782.42	326.99	25.1
C-911	3,910,716.79	11,685,941.76	286.69	15.3
C-912	3,909,959.42	11,686,349.77	271.16	2.8
C-913	3,910,999.95	11,686,812.54	268.65	20.0
C-914	3,910,360.20	11,688,917.62	267.86	31.0
C-915	3,909,784.60	11,686,794.40	320.92	54.0
C-916	3,909,584.68	11,686,372.70	312.91	49.1
C-917	3,909,337.29	11,686,293.79	320.37	49.2
C-918	3,909,151.49	11,685,509.11	329.55	25.1
C-919	3,909,154.30	11,685,255.41	338.06	25.1
C-920	3,909,071.70	11,685,870.40	324.73	25.1
C-921	3,910,112.20	11,685,717.17	281.10	30.0
C-922	3,909,889.28	11,684,055.95	311.73	20.3
C-923	3,910,107.49	11,683,828.42	283.03	22.2

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Table 2.5-208 Elevation, Depth, and Thickness of the Subsurface Zones

Boring Number	Top Elevation of Zones (ft)						Top Depth of Zones (ft)						Thickness of Zones (ft)				
	I	IIA	IIB	III	III-IV	IV	I	IIA	IIB	III	III-IV	IV	I	IIA	IIB	III	III-IV
B-901	309.4	309.4	279.9	269.5	229.4	174.4	0.0	0.0	29.5	39.9	80.0	135.0	0.0	29.5	10.4	40.1	55.0
B-902	302.2	302.2	283.0	283.0	-	278.4	0.0	0.0	19.2	19.2	-	23.8	0.0	19.2	0.0	4.6	-
B-903	301.6	301.6	281.9	279.0	220.8	185.6	0.0	0.0	19.7	22.6	80.8	116.0	0.0	19.7	2.9	58.2	35.2
B-904	316.8	316.8	288.3	270.0	235.1	195.1	0.0	0.0	28.5	46.8	81.7	121.7	0.0	28.5	18.3	34.9	40.0
B-905	306.7	306.7	286.8	282.9	271.4	176.2	0.0	0.0	19.9	23.8	35.3	130.5	0.0	19.9	3.9	11.5	95.2
B-906	311.7	311.7	282.8	276.8	262.0	176.2	0.0	0.0	28.9	34.9	49.7	135.5	0.0	28.9	6.0	14.8	85.8
B-907	322.7	322.7	287.7	283.7	207.2	177.2	0.0	0.0	35.0	39.0	115.5	145.5	0.0	35.0	4.0	76.5	30.0
B-908	307.7	307.7	280.7	245.0	-	241.3	0.0	0.0	27.0	62.7	-	66.4	0.0	27.0	35.7	3.7	-
B-909	304.9	304.9	275.9	248.0	-	223.0	0.0	0.0	29.0	56.9	-	81.9	0.0	29.0	27.9	25.0	-
B-910	316.5	316.5	294.5	274.5	226.1	-	0.0	0.0	22.0	42.0	90.4	-	0.0	22.0	20.0	48.4	-
B-911	299.8	299.8	282.8	278.8	268.7	233.8	0.0	0.0	17.0	21.0	31.1	66.0	0.0	17.0	4.0	10.1	34.9
B-911A	299.9	299.9	282.9	278.8	268.7	233.8	0.0	0.0	17.0	21.1	31.2	66.1	0.0	17.0	4.1	10.1	34.9
B-912	275.1	275.1	255.5	251.0	-	238.3	0.0	0.0	19.6	24.1	-	36.8	0.0	19.6	4.5	12.7	-
B-913	273.4	273.4	223.4	217.9	-	215.5	0.0	0.0	50.0	55.5	-	57.9	0.0	50.0	5.5	2.4	-
B-914	297.4	297.4	275.4	275.4	236.9	202.1	0.0	0.0	22.0	22.0	60.5	95.3	0.0	22.0	0.0	38.5	34.8
B-915	301.8	301.8	288.3	284.8	279.4	-	0.0	0.0	13.5	17.0	22.4	-	0.0	13.5	3.5	5.4	-
B-916	276.2	276.2	251.1	-	-	250.6	0.0	0.0	25.1	-	-	25.6	0.0	25.1	0.5	-	-
B-917	274.9	274.9	217.9	206.4	187.1	178.8	0.0	0.0	57.0	68.5	87.8	96.1	0.0	57.0	11.5	19.3	8.3
B-918	272.1	271.1	267.0	245.8	-	239.8	0.0	1.0	5.1	26.3	-	32.3	1.0	4.1	21.2	6.0	-
B-919	317.8	317.8	294.8	279.9	264.7	-	0.0	0.0	23.0	37.9	53.1	-	0.0	23.0	14.9	15.2	-

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Table 2.5-208 Elevation, Depth, and Thickness of the Subsurface Zones

Boring Number	Top Elevation of Zones (ft)						Top Depth of Zones (ft)						Thickness of Zones (ft)				
	I	IIA	IIB	III	III-IV	IV	I	IIA	IIB	III	III-IV	IV	I	IIA	IIB	III	III-IV
B-920	327.2	324.8	289.2	274.2	-	221.5	0.0	2.4	38.0	53.0	-	105.7	2.4	35.6	15.0	52.7	-
B-921	308.0	308.0	260.0	236.2	-	-	0.0	0.0	48.0	71.8	-	-	0.0	48.0	23.8	-	-
B-921A	307.4	307.4	259.4	236.2	-	-	0.0	0.0	48.0	71.2	-	-	0.0	48.0	23.2	-	-
B-922	271.3	271.3	265.0	262.5	257.3	-	0.0	0.0	6.3	8.8	14.0	-	0.0	6.3	2.5	5.2	-
B-922A	271.3	271.3	271.3	263.1	254.8	209.8	0.0	0.0	0.0	8.2	16.5	61.5	0.0	0.0	8.2	8.3	45.0
B-923	272.0	269.2	266.8	-	266.8	260.3	0.0	2.8	5.2	-	5.2	11.7	2.8	2.4	0.0	-	6.5
B-924	271.5	271.1	265.0	265.0	252.9	227.9	0.0	0.4	6.5	6.5	18.6	43.6	0.4	6.1	0.0	12.1	25.0
B-925	270.0	270.0	253.0	-	249.6	213.7	0.0	0.0	17.0	-	20.4	56.3	0.0	17.0	3.4	-	35.9
B-926	289.0	289.0	235.0	235.0	225.2	179.5	0.0	0.0	54.0	54.0	63.8	109.5	0.0	54.0	0.0	9.8	45.7
B-927	292.5	292.5	268.5	-	252.7	217.9	0.0	0.0	24.0	-	39.8	74.6	0.0	24.0	15.8	-	34.8
B-928	272.2	272.2	244.2	235.1	220.1	212.0	0.0	0.0	28.0	37.1	52.1	60.2	0.0	28.0	9.1	15.0	8.1
B-928A	271.8	271.8	243.8	235.1	220.1	212.0	0.0	0.0	28.0	36.7	51.7	59.8	0.0	28.0	8.7	15.0	8.1
B-929	329.0	329.0	283.0	265.0	-	-	0.0	0.0	46.0	64.0	-	-	0.0	46.0	18.0	-	-
B-929A	329.0	329.0	283.0	265.0	-	-	0.0	0.0	46.0	64.0	-	-	0.0	46.0	18.0	-	-
B-930	326.1	323.7	265.1	244.1	-	-	0.0	2.4	61.0	82.0	-	-	2.4	58.6	21.0	-	-
B-931	278.5	278.5	228.7	221.5	-	-	0.0	0.0	49.8	57.0	-	-	0.0	49.8	7.2	-	-
B-932	249.9	231.9	221.9	-	-	-	0.0	18.0	28.0	-	-	-	18.0	10.0	-	-	-
B-933	296.5	291.0	274.5	269.5	248.3	239.6	0.0	5.5	22.0	27.0	48.2	56.9	5.5	16.5	5.0	21.2	8.7
B-933A	296.6	291.1	274.6	269.5	248.3	239.6	0.0	5.5	22.0	27.1	48.3	57.0	5.5	16.5	5.1	21.2	8.7
B-934	294.8	294.8	252.8	252.8	-	246.4	0.0	0.0	42.0	42.0	-	48.4	0.0	42.0	0.0	6.4	-

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Table 2.5-208 Elevation, Depth, and Thickness of the Subsurface Zones

Boring Number	Top Elevation of Zones (ft)						Top Depth of Zones (ft)						Thickness of Zones (ft)				
	I	IIA	IIB	III	III-IV	IV	I	IIA	IIB	III	III-IV	IV	I	IIA	IIB	III	III-IV
B-936	286.6	286.6	266.3	253.1	190.6	-	0.0	0.0	20.3	33.5	96.0	-	0.0	20.3	13.2	62.5	-
B-937	270.3	270.3	245.3	237.0	220.0	-	0.0	0.0	25.0	33.3	50.3	-	0.0	25.0	8.3	17.0	-
B-939	254.0	254.0	215.2	-	-	-	0.0	0.0	38.8	-	-	-	0.0	38.8	-	-	-
B-940	268.3	268.3	249.8	249.8	212.1	-	0.0	0.0	18.5	18.5	56.2	-	0.0	18.5	0.0	37.7	-
B-941	267.2	267.2	258.7	219.3	-	205.8	0.0	0.0	8.5	47.9	-	61.4	0.0	8.5	39.4	13.5	-
B-942	291.8	291.8	285.8	-	-	263.0	0.0	0.0	6.0	-	-	28.8	0.0	6.0	22.8	-	-
B-943	300.4	300.4	283.9	278.0	268.5	220.1	0.0	0.0	16.5	22.4	31.9	80.3	0.0	16.5	5.9	9.5	48.4
B-944	334.7	334.7	299.7	-	281.2	-	0.0	0.0	35.0	-	53.5	-	0.0	35.0	18.5	-	-
B-945	281.5	281.5	228.1	-	221.0	210.6	0.0	0.0	53.4	-	60.5	70.9	0.0	53.4	7.1	-	10.4
B-946	333.4	333.4	301.2	-	291.6	-	0.0	0.0	32.2	-	41.8	-	0.0	32.2	9.6	-	-
B-947	312.5	312.5	260.8	248.8	-	-	0.0	0.0	51.7	63.7	-	-	0.0	51.7	12.0	-	-
B-948	310.4	310.4	288.4	281.9	274.7	-	0.0	0.0	22.0	28.5	35.7	-	0.0	22.0	6.5	7.2	-
B-949	334.8	334.8	281.9	-	258.4	-	0.0	0.0	52.9	-	76.4	-	0.0	52.9	23.5	-	-
B-950	282.5	282.5	261.8	-	232.2	218.7	0.0	0.0	20.7	-	50.3	63.8	0.0	20.7	29.6	-	13.5
B-951	249.9	249.9	230.4	209.3	-	179.9	0.0	0.0	19.5	40.6	-	70.0	0.0	19.5	21.1	29.4	-

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Table 2.5-209 Type and Number of Laboratory Tests Performed

Material	Test	Number
Soil	Natural moisture content	111
	Specific gravity	6
	Sieve and hydrometer analysis	52
	Grain size analysis with no. 200 wash	64
	Atterberg limits	18
	Chemical analysis (pH, chloride, sulfate)	20
	Triaxial consolidated-undrained compression	6
	Resonant column torsional shear	5
	California bearing ratio	5
	Moisture density (modified Proctor)	9
Rock	Unit weight	82
	Unconfined compression	55
	Unconfined compression with stress-strain measurements	27

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Table 2.5-210 Results of Laboratory Tests on Soil Samples

Boring Number	Sample Number	Depth (ft)	Sample Type	Gravel [1] (%)	Sand [1] (%)	Fines [2] (%)	Silt [1] (%)	0.005 mm Clay [1] (%)	USCS Symbol	Natural Moisture (%)	LL	PI	G_s	pH [3]	Chloride (mg/kg) [3], [6], [7]	Sulfate (mg/kg) [3], [6], [7]
B-901	B-901-2	3.5-5.0	SPT	0.0	53.6	46.4	10.8	35.6	[8]	21.5	[8]	[8]	[8]	[8]	[8]	[8]
B-901	B-901-4	11.5-13.0	SPT	0.0	76.6	23.4	16.0	7.4		10.2				5.8	ND [5]	ND [5]
B-901	B-901-6	22.2-23.7	SPT	0.0	76.8	23.2				16.4						
B-901	B-901-9	37.2-38.7	SPT	0.7	71.9	22.5	15.2	7.3		16.4						
B-901	UD-2	9.5-11.5 [4]	UD	0.0	78.0	22.0	12.6	9.4		15.0						
B-902	B-902-2	3.5-5.0	SPT	0.0	86.1	13.9				5.6						
B-902	B-902-4	8.5-10.0	SPT	1.3	71.0	29.0	13.4	15.6	SM	23.9	33	7				
B-902	B-902-6	13.5-15.0	SPT	0.0	80.0	20.0				14.0						
B-907	B-907-2	3.5-5.0	SPT	0.0	67.0	33.0	17.7	15.3	SM	14.0	33	8				
B-907	B-907-3	5.5-7.0	SPT	0.0	74.9	25.1				16.4				4.8	51.1 J	ND [5]
B-907	B-907-5	11.0-12.5	SPT	0.0	76.0	24.0				20.2						
B-907	B-907-7	17.5-19.0	SPT	0.0	80.9	19.1	11.7	7.4		12.3						
B-907	B-907-9	27.5-29.0	SPT	0.0	73.9	26.1										
B-907	B-907-10	32.5-34.0	SPT	0.0	66.6	23.4										
B-908	B-908-3	6.0-7.5	SPT	2.0	72.6	25.4	11.6	13.8		12.3			2.62			
B-908	B-908-6	13.5-15.0	SPT	0.0	76.6	23.4							2.69			
B-908	B-908-8	23.7-25.2	SPT	0.0	68.1	31.9										
B-908	B-908-13	47.1-48.6	SPT	0.0	76.0	24.0	18.9	5.1		14.5						
B-909	B-909-3	6.0-7.5	SPT	0.0	66.9	33.1	19.3	13.8	SM	25.9	57	12				

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Table 2.5-210 Results of Laboratory Tests on Soil Samples

Boring Number	Sample Number	Depth (ft)	Sample Type	Gravel (1) (%)	Sand (1) (%)	Fines (2) (%)	Silt (1) (%)	0.005 mm Clay (1) (%)	USCS Symbol	Natural Moisture (%)	LL	PI	G_s	pH (3)	Chloride (mg/kg) (3), (6), (7)	Sulfate (mg/kg) (3), (6), (7)
B-909	B-909-5	11.0-12.5	SPT	0.0	77.6	22.4				31.4				5.4	137 [J]	6.7
B-909	B-909-7	18.5-20.0	SPT	0.0	63.7	36.3	29.0	7.3	SM	25.1	30	4				
B-909	B-909-8	23.5-25.0	SPT	1.7	56.1	42.2				35.4						
B-909	B-909-12	41.9-43.4	SPT	0.0	75.3	24.7				17.6						
B-910	B-910-2	3.5-5.0	SPT	4.0	31.9	64.1	12.1	52.0		27.7						
B-910	B-910-5	11.0-12.5	SPT							30.5	45	13		5.8	3.6 [J]	5.1 [B]
B-910	B-910-7	18.5-20.0	SPT	0.0	46.4	53.6	43.1	10.5		33.1						
B-910	B-910-9	25.9-27.4	SPT	2.3	76.3	21.4				14.6				6.7	5.2 [J]	4.2 [B]
B-911	B-911-2	3.5-5.0	SPT	0.3	59.1	40.6				12.8						
B-911	B-911-4	8.0-9.5	SPT	0.0	70.6	29.4	13.6	15.8		19.6						
B-911	B-911-5	11.0-12.5	SPT	0.0	78.3	21.7								5.6	3.4 [J]	ND (5)
B-911	B-911-7	18.5-20.0	SPT	0.1	80.0	19.9				11.1						
B-912	B-912-1	9.1-10.6	SPT	0.0	73.7	26.3	20.8	5.5		24.0						
B-912	B-912-3	14.1-15.6	SPT	0.0	72.6	27.4				15.2						
B-912	B-912-4	19.1-19.9	SPT	14.5	84.9	0.6				15.7						
B-913	B-913-8	43.5-48.5	SPT	0.0	72.3	27.7										
B-914	B-914-2	3.5-5.0	SPT	0.1	52.9	47.0	21.0	26.0	SC	16.6	27	10				
B-914	B-914-3	6.0-7.5	SPT	4.0	63.0	33.0										
B-914	B-914-5	11.0-13.5	SPT	2.1	78.0	19.9										

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Table 2.5-210 Results of Laboratory Tests on Soil Samples

Boring Number	Sample Number	Depth (ft)	Sample Type	Gravel (1) (%)	Sand (1) (%)	Fines (2) (%)	Silt (1) (%)	0.005 mm Clay (1) (%)	USCS Symbol	Natural Moisture (%)	LL	PI	G_s	pH (3)	Chloride (mg/kg) (3), (6), (7)	Sulfate (mg/kg) (3), (6), (7)
B-914	B-914-7	19.0-20.5	SPT	27.8	61.0	11.2	8.6	2.6		20.8						
B-914	B-914-9	35.6-37.1	SPT	5.7	70.1	24.2								6.8	8.4 [J]	ND [(5)]
B-914	B-914-10	40.6-42.1	SPT	0.1	74.4	25.5	19.5	6.0		20.5						
B-917	B-917-13	48.5-53.5	SPT	0.0	81.9	18.1	15.0	3.1								
B-918	B-918-2	1.8-3.2	SPT	1.2	85.7	13.1	7.3	5.8		15.8			2.68			
B-918	B-918-3	5.1-6.6	SPT	0.0	85.0	15.0				13.3				6.9	8.0 [J]	9.4
B-918	B-918-4	9.3-10.8	SPT	0.0	80.6	19.4	13.4	6.0		13.7						
B-918	B-918-6	13.2-14.7	SPT	0.0	77.7	22.3				13.9						
B-918	B-918-8	22.4-23.9	SPT	1.4	79.4	19.2				17.8						
B-919	B-919-1	1.5-3.0	SPT							18.6	32	11				
B-919	B-919-3	5.9-7.4	SPT	2.5	80.9	16.6				11.1						
B-919	B-919-5	11.0-12.5	SPT	0.6	80.4	19.0				11.2						
B-919	B-919-7	18.9-19.4	SPT	3.7	75.5	20.8	10.8	10.0		13.8						
B-919	B-919-13	51.3-52.8	SPT	0.0	65.9	34.1	26.0	8.1		17.9						
B-920	B-920-1	2.0-3.5	SPT							25.2						
B-920	B-920-2	3.8-5.3	SPT											5.9	1.5 [B J]	7.5
B-920	B-920-3	6.0-7.5	SPT	0.3	58.9	40.8				24.1						
B-920	B-920-6	13.8-15.3	SPT							15.7				6.5	63.0 [J]	7.5
B-920	B-920-7	18.8-20.3	SPT	0.0	72.3	27.7	21.3	6.4		15.4						

Table 2.5-210 Results of Laboratory Tests on Soil Samples

Boring Number	Sample Number	Depth (ft)	Sample Type	Gravel (1) (%)	Sand (1) (%)	Fines (2) (%)	Silt (1) (%)	0.005 mm Clay (1) (%)	USCS Symbol	Natural Moisture (%)	LL	PI	G_s	pH (3)	Chloride (mg/kg) (3), (6), (7)	Sulfate (mg/kg) (3), (6), (7)
B-920	B-920-9	27.3-28.8	SPT	0.0	79.9	20.1				19.5						
B-920	B-920-12	43.5-44.7	SPT							12.9				6.9	1.4 [B J]	2.3 [B]
B-921	B-921-1	1.5-3.0	SPT	11.5	52.1	36.4				12.0						
B-921	B-921-3	6.0-7.5	SPT	0.0	41.3	58.7	29.2	29.5	CL	24.8	34	14				
B-921	B-921-4	8.5-10.0	SPT	0.0	53.5	46.5	37.3	9.2		28.0				7.0	4.4 [J]	10.8
B-921	B-921-6	13.5-15.0	SPT	0.0	74.2	25.8	16.1	9.7		26.0						
B-921	B-921-8	23.8-25.3	SPT							32.1	38	NP				
B-921	B-921-10	33.8-35.3	SPT	0.0	75.5	24.5				20.4						
B-921	B-921-11	38.8-40.3	SPT	0.0	81.3	18.7				15.8						
B-921	B-921-16	63.8-65.3	SPT	0.0	75.1	24.9	18.2	6.7		8.5						
B-923	B-923-2	3.3-4.8	SPT	10.9	55.5	33.6	16.7	16.9	SC	22.5	33	10				
B-924	B-924-2	3.5-5.0	SPT	23.2	65.8	11.0	7.9	3.1		2.1						
B-924	B-924-3	6.0-7.5	SPT	11.1	74.5	14.4				4.8						
B-927	B-927-1	1.5-3.0	SPT	0.0	61.4	38.6	12.6	26.0	SC	14.1	28	10				
B-927	B-927-2	3.5-5.0	SPT	0.0	75.8	24.2				11.7						
B-927	B-927-3	6.0-7.5	SPT	0.0	73.2	26.8	17.1	9.7		12.2						
B-927	B-927-4	8.5-10.0	SPT	0.0	83.3	16.7				6.8				5.8	2.8 [J]	4.3 [B]
B-927	B-927-6	13.5-15.0	SPT	0.0	81.2	18.8				11.2						
B-927	B-927-7	18.5-20.0	SPT	0.0	76.2	23.8				11.4						

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Table 2.5-210 Results of Laboratory Tests on Soil Samples

Boring Number	Sample Number	Depth (ft)	Sample Type	Gravel (1) (%)	Sand (1) (%)	Fines (2) (%)	Silt (1) (%)	0.005 mm Clay (1) (%)	USCS Symbol	Natural Moisture (%)	LL	PI	G_s	pH (3)	Chloride (mg/kg) (3), (6), (7)	Sulfate (mg/kg) (3), (6), (7)
B-927	B-927-8	23.5-25.0	SPT	0.0	79.7	20.3				15.7				7.4	5.6 [J]	3.4 [B]
B-928	B-928-2	3.5-5.0	SPT	0.0	78.4	21.6				17.9						
B-928	B-928-4	8.3-9.8	SPT	0.0	73.4	26.6				18.5				6.8	120.0 [J]	4.9 [B]
B-928	B-928-6	14.0-15.5	SPT	0.0	77.0	23.0	17.8	5.2		24.5						
B-928	B-928-8	22.1-23.6	SPT	0.0	78.7	21.3				17.0						
B-928	B-928-9	27.1-28.6	SPT	0.0	74.7	25.3	19.2	6.1		16.4						
B-928 A	UD-3	20-22 (4)	UD	0.0	82.0	18.0	13.2	4.8								
B-929	B-929-1	1.5-3.0	SPT	12.2	43.7	44.1	16.6	27.5	SC	14.5	36	17				
B-929	B-929-2	3.5-5.0	SPT								54	16				
B-929	B-929-4	8.7-10.2	SPT	0.0	65.5	34.5				18.9				5.9	2.8 [J]	2.7 [B]
B-929	B-929-5	13.5-15.0	SPT	0.0	73.8	26.2				19.6						
B-929	B-929-7	23.0-24.5	SPT	0.0	76.9	23.1	17.0	6.1		18.8						
B-929	B-929-9	33.0-34.5	SPT	0.0	82.7	17.3				16.9						
B-929	B-929-11	43.0-44.5	SPT	0.7	81.4	17.9				17.2						
B-929	B-929-13	53.0-54.5	SPT	0.0	80.0	20.0				13.8						
B-929A	UD-1	15.0-16.8 (4)	UD	0.0	78.6	21.4	15.1	6.3		13.1						
B 929A	UD-6	40-41.8 (4)	UD	0.0	83.3	16.7	11.7	5.0		16.9						
B-931	B-931-10	47.3-48.8	SPT	0.0	78.5	21.5	15.9	5.6								
B-932	B-932-5	19.0-20.5	SPT	0.0	77.7	22.3	15.7	6.6		21.5						

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Table 2.5-210 Results of Laboratory Tests on Soil Samples

Boring Number	Sample Number	Depth (ft)	Sample Type	Gravel [1] (%)	Sand [1] (%)	Fines [2] (%)	Silt [1] (%)	0.005 mm Clay [1] (%)	USCS Symbol	Natural Moisture (%)	LL	PI	G_s	pH (3)	Chloride (mg/kg) (3), (6), (7)	Sulfate (mg/kg) (3), (6), (7)
B-933	B-933-3	6.0-7.5	SPT	0.0	62.3	37.7	22.6	15.1	SM	24.2	28	3				
B-933	B-933-5	11.2-12.7	SPT	0.0	58.8	41.2				25.9				5.4	210 [J]	3.0 [B]
B-933	B-933-7	19.5-21.0	SPT	0.0	76.6	23.4				26.7						
B-933	B-933-8	24.5-25.0	SPT	0.0	80.5	19.5				18.7						
B-945	B-945-1	1.5-3.0	SPT	0.0	82.0	18.0				14.5						
B-945	B-945-3	4.7-6.2	SPT	0.0	75.5	24.5	16.2	8.3		15.9						
B-945	B-945-5	11.3-12.8	SPT	0.0	84.2	15.8				21.6				6.4	6.9 [J]	3.1 [B]
B-945	B-945-7	19.4-20.9	SPT	0.0	84.8	15.2				27.6			2.58			
B-945	B-945-9	27.8-29.4	SPT	0.0	82.9	17.1	10.2	6.9		24.1						
B-945	B-945-11	39.4-40.9	SPT	0.0	90.1	9.9				20.4						
B-945	B-945-13	49.4-50.9	SPT	0.0	90.3	9.7				15.6						
B-947	B-947-1	1.5-3.0	SPT							16.7	55	25	2.60			
B-947	B-947-3	4.5-6.0	SPT	0.0	38.3	61.7	23.5	38.2	MH	36.0	56	19				
B-947	B-947-4	8.5-10.0	SPT	0.0	60.0	40.0			SM	20.7	38	9				
B-947	B-947-5	9.5-11.0	SPT	1.6	55.9	42.5	21.1	21.4		28.2			2.78			
B-947	B-947-6	13.5-15.0	SPT	0.0	30.5	69.5				22.5						
B-947	B-947-7	17.2-18.7	SPT	0.0	75.8	24.2				21.1				6.4	21.4 [J]	6.4
B-947	B-947-8	22.2-23.7	SPT	0.6	79.4	20.0	10.7	9.3		24.3						
B-947	B-947-9	28.7-30.2	SPT	0.0	66.6	33.4				28.8	33	NP				

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Table 2.5-210 Results of Laboratory Tests on Soil Samples

Boring Number	Sample Number	Depth (ft)	Sample Type	Gravel (1) (%)	Sand (1) (%)	Fines (2) (%)	Silt (1) (%)	0.005 mm Clay (1) (%)	USCS Symbol	Natural Moisture (%)	LL	PI	G_s	pH (3)	Chloride (mg/kg) (3), (6), (7)	Sulfate (mg/kg) (3), (6), (7)
B-947	B-947-10	33.7-35.2	SPT	0.0	81.3	18.7				20.2						
B-947	B-947-11	38.7-40.2	SPT	0.0	85.8	14.2				16.9						
B-947	B-947-12	42.2-43.7	SPT	0.0	79.7	20.3	13.4	6.9		20.5						
B-948	B-948-1	1.5-3.0	SPT	0.0	54.7	45.3				83.7						
B-948	B-948-3	6.0-7.5	SPT	0.0	51.1	48.9				16.2				5.7	3.8 J	ND (5)
B-948	B-948-5	9.5-11.0	SPT	0.0	31.0	69.0	61.9	7.1		13.7						
B-948	B-948-7	18.5-20.0	SPT	0.0	35.9	64.1				15.2						
B-948	B-948-8	23.5-24.4	SPT	0.0	77.7	22.3				13.6						
B-951	B-951-8	23.0-24.5	SPT	0.2	82.9	16.9	10.5	6.4		13.9						

(1) Due to computer roundoff, particle size fractions may total 100 ±1. Fines include silt plus clay.

(2) Fines include silt plus clay.

(3) Tests performed by STL - St. Louis, MO

(4) Depth interval shown reflects total pushed depth of UD tube.

(5) ND indicates analyte not detected at or above the Method Detection Limit

(6) B = Estimated Result. Analyte detected above the Method Detection Limit but not above the Reporting Limit.

(7) J = Method blank contamination. The associated method blank contains the target analyte at a reportable level

(8) Shaded cells indicate that information not obtained.

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Table 2.5-210 Results of Laboratory Tests on Soil Samples; Consolidated-Undrained Triaxial Tests

Source of Sample	Sample No.	Sample Depth [1] (ft)	Sample Type	Test Type	C' (psf)	Φ' (degree)	C (psf)	Φ' (degree)	Comment
B-901	UD-2	9.5-11.5	UD Tube	CU	0.0	33.6	0.0	37.5	
B-928 A	UD-3	20-22	UD Tube	CU	423.4	31.4	103.7	41.2	
B-929 A	UD-1	15-16.75	UD Tube	CU	5.4	32.4	178.6	35.8	Only 2 points tested due to limited sample
B-929 A	UD-4	30-31.5	UD Tube	CU	0.0	33.0	0.0	33.0	Only 2 points tested due to limited sample
B-929 A	UD-6	40-41.5	UD Tube	CU	0.0	36.1	318.2	36.4	
B-933 A	UD-2	15-16.25	UD Tube	CU	55.0	32.6	479.5	30.5	Only 2 points tested due to limited sample

(1) Sample depth shown reflects the depth of start of push plus the length of the recovered sample

NAPS COL 2.0-29-A **Table 2.5-210 Results of Laboratory Tests on Soil Samples Moisture-Density and CBR Tests**

		Moisture/Density Results [A]			CBR Results [B]			
Source of Sample	**Sample No.**	**Natural Moisture (%)**	**Maximum Dry Density (pcf)**	**Optimum Moisture (%)**	**Molded Density (pcf)**	**Molded Moisture (%)**	**Soaked CBR (0.10") (%)**	**Soaked CBR (0.20") (%)**
Test Pit 1	TP-1-1	23.4	108.7	17.6	Not Tested			
Test Pit 1	TP-1-2	22.6	108.8	17.1	90.3	17.0	1.2	1.6
					94.4	17.0	6.3	5.5
					105.3	17.2	14.7	15.6
Test Pit 2	TP-2	22.6	100.4	22.3	83.0	22.8	1.1	1.1
					89.1	22.0	1.3	1.2
					101.0	22.0	6.2	6.5
Test Pit 3	TP 3-1	16.1	124.9	9.5	Not Tested			
Test Pit 3	TP 3-2	12.4	124.5	10.9	117.5	10.7	5.9	6.0
					122.9	10.6	3.2	5.0
					125.6	10.5	4.2	8.4
Test Pit 4	TP 4-1	30.2	108.6	17.1	Not Tested			
Test Pit 4 [C]	TP 4-2	15.2	125.5	10.8	119.4	11.0	4.9	7.3
					121.5	10.6	8.8	11.9
Test Pit 5	TP 5	9.4	126	9.2	Not Tested			
Test Pit 6	TP 6	18.2	116.1	13.2	110.3	12.3	6.9	8.0
					111.7	12.7	6.4	9.5
					115.1	12.3	12.1	13.8

A Proctor Test results, ASTM D 1557-02 Method A Modified

B California Bearing Ratio Test results, ASTM D 1883-05 (Section 7.12)

C Insufficient Material for three tests

NAPS COL 2.0-29-A

Table 2.5-211 Results of Unconfined Compression Tests on Rock

Boring No.	Run Number	Sample Top Depth (ft)	Sample Length (L) (Inches)	Sample Diameter (D) (inches)	L/D Ratio	Unit Weight (pcf)	Type of Break [1]	Unconfined Compressive Strength (psi)[2]	Young's Modulus (psi)	Poisson's Ratio
B-901	5	54.0	5.27	2.49	2.1	160	Shear	4,375	(ND) [3]	(ND)
B-901	7	60.3	5.27	2.49	2.1	162	Columnar	15,425	3,970,000	* [4]
B-901	14	97.9	5.34	2.50	2.1	162	C&S	12,629	(ND)	(ND)
B-901	25	129.5	5.35	2.49	2.1	164	C&S	14,171	(ND)	(ND)
B-901	34	170.5	5.33	2.40	2.2	168	Shear	10,865	5,360,000	0.31
B-901	42	208.5	5.32	2.40	2.2	163	Shear	12,777	(ND)	(ND)
B-901	51	240.5	5.35	2.39	2.2	165	C&S	23,619	(ND)	(ND)
B-901	59	280.5	5.36	2.39	2.2	164	C&S	25,335	8,320,000	0.39
B-902	3	27.3	5.29	2.38	2.2	162	C&S	14,947	4,090,000	* [4]
B-902	9	47.4	5.35	2.40	2.2	163	Shear	21,007	(ND)	(ND)
B-902	14	72.3	5.34	2.40	2.2	164	C&S	25,100	(ND)	(ND)
B-902	18	92.8	5.32	2.40	2.2	164	Shear	6,030	1,840,000	0.42
B-902	28	141.9	5.31	2.40	2.2	170	Shear	6,982	(ND)	(ND)
B-902	38	184.6	5.36	2.40	2.2	163	C&S	27,303	(ND)	(ND)
B-907	3	51.9	5.29	2.45	2.2	152	Shear	957	(ND)	(ND)
B-907	12	90.0	5.23	2.46	2.1	155	Shear	751	(ND)	(ND)
B-907	24	116.8	5.27	2.47	2.1	173	Shear	4,599	(ND)	(ND)
B-907	27	131.8	5.32	2.48	2.1	173	C&S	8,519	(ND)	(ND)
B-907	33	160.8	5.32	2.50	2.1	163	Columnar	19,333	7,700,000	0.30

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Table 2.5-211 Results of Unconfined Compression Tests on Rock

Boring No.	Run Number	Sample Top Depth (ft)	Sample Length (L) (Inches)	Sample Diameter (D) (inches)	L/D Ratio	Unit Weight (pcf)	Type of Break (1)	Unconfined Compressive Strength (psi)(2)	Young's Modulus (psi)	Poisson's Ratio
B-907	40	200.0	5.35	2.50	2.1	165	C&S	20,166	(ND)	(ND)
B-908	2	67.5	5.32	2.38	2.2	163	Shear	5,476	(ND) 3	(ND)
B-908	4	79.4	5.25	2.39	2.2	164	C&S	14,695	3,400,000	0.41
B-908	7	96.0	5.31	2.39	2.2	163	Shear	17,164	(ND)	(ND)
B-908	11	112.7	5.32	2.38	2.2	178	Shear	15,284	(ND)	(ND)
B-908	17	135.7	5.28	2.38	2.2	187	Shear	5,670	3,180,000	0.21
B-908	20	146.8	5.31	2.38	2.2	173	Shear	7,687	(ND)	(ND)
B-909	11	82.4	5.32	2.39	2.2	176	C&S	9,464	3,520,000	* (4)
B-909	14	96.5	5.28	2.39	2.2	190	Shear	5,897	(ND)	(ND)
B-909	17	107.4	5.35	2.39	2.2	179	Shear	3,938	(ND)	(ND)
B-909	21	127.4	5.35	2.39	2.2	174	Shear	8,167	(ND)	(ND)
B-909	26	152.3	5.27	2.38	2.2	184	C&S	6,467	4,600,000	0.39
B-909	33	187.3	5.32	2.39	2.2	175	Shear	9,305	(ND)	(ND)
B-910	5	53.1	5.27	2.15	2.2	159	Shear	6,935	(ND)	(ND)
B-910	13	91.1	5.24	2.15	2.2	159	Shear	4,821	670,000	* (4)
B-910	20	120.9	5.27	2.40	2.2	163	Columnar	9,395	(ND)	(ND)
B-910	24	142.1	5.35	2.40	2.2	168	C&S	28,834	(ND)	(ND)
B-911	3	34.3	5.27	2.37	2.2	161	Shear	5,558	1,230,000	* (4)
B-911	5	44.3	5.28	2.38	2.2	162	Cone	10,209	(ND)	(ND)

NAPS COL 2.0-29-A

Table 2.5-211 Results of Unconfined Compression Tests on Rock

Boring No.	Run Number	Sample Top Depth (ft)	Sample Length (L) (Inches)	Sample Diameter (D) (inches)	L/D Ratio	Unit Weight (pcf)	Type of Break [1]	Unconfined Compressive Strength (psi)[2]	Young's Modulus (psi)	Poisson's Ratio
B-911	10	66.5	5.35	2.39	2.2	164	Cone	24,646	(ND)	(ND)
B-911	13	82.1	5.36	2.40	2.2	164	C&S	20,431	5,730,000	0.40
B-911	16	97.6	5.36	2.40	2.2	163	Shear	6,561	(ND) [3]	(ND)
B-912	3	37.1	5.32	2.39	2.2	170	C&S	3,524	2,570,000	(ND)
B-912	5	48.9	5.26	2.40	2.2	163	C&S	12,992	(ND)	(ND)
B-912	8	62.2	5.26	2.40	2.2	164	C&S	32,680	(ND)	(ND)
B-912	12	82.4	5.25	2.40	2.2	163	Shear	27,356	(ND)	(ND)
B-912	17	111.4	5.32	2.40	2.2	163	Shear	16,702	8,220,000	0.31
B-912	24	143.9	5.26	2.40	2.2	161	Columnar	15,996	(ND)	(ND)
B-914	8	63.8	5.34	2.40	2.2	169	Cone	17,866	(ND)	(ND)
B-914	10	75.3	5.32	2.40	2.2	164	C&S	36,600	(ND)	(ND)
B-914	15	95.8	5.37	2.40	2.2	164	C&S	29,776	8,980,000	0.31
B-914	20	120.6	5.32	2.39	2.2	169	C&S	17,942	(ND)	(ND)
B-914	26	151.4	5.31	2.40	2.2	166	C&S	16,517	8,930,000	0.32
B-914	34	192.7	5.32	2.40	2.2	163	Cone	30,162	(ND)	(ND)
B-918	2	31.7	5.29	2.39	2.2	164	Shear	19,038	(ND)	(ND)
B-918	4	37.1	5.32	2.40	2.2	164	C&S	29,636	9,530,000	0.35
B-918	7	51.6	5.29	2.40	2.2	165	Cone	15,409	(ND)	(ND)
B-918	9	60.7	5.32	2.40	2.2	164	Columnar	21,064	(ND)	(ND)

NAPS COL 2.0-29-A

Table 2.5-211 Results of Unconfined Compression Tests on Rock

Boring No.	Run Number	Sample Top Depth (ft)	Sample Length (L) (Inches)	Sample Diameter (D) (inches)	L/D Ratio	Unit Weight (pcf)	Type of Break (1)	Unconfined Compressive Strength (psi)(2)	Young's Modulus (psi)	Poisson's Ratio
B-918	15	88.1	5.28	2.40	2.2	165	Shear	21,944	7,850,000	0.24
B-918	22	122.0	5.25	2.40	2.2	166	C&S	33,610	(ND)	(ND)
B-920	7	90.2	5.28	2.39	2.2	160	Shear	1,021	(ND)	(ND)
B-920	11	107.7	5.32	2.39	2.2	163	Cone	29,621	8,500,000	0.34
B-920	13	119.1	5.33	2.39	2.2	181	Shear	9,456	(ND)	(ND)
B-920	18	141.1	5.35	2.40	2.2	166	Cone	18,040	5,970,000	* (4)
B-923	6	20.0	5.35	2.39	2.2	164	C&S	28,911	8,510,000	0.28
B-923	9	30.8	5.35	2.39	2.2	162	C&S	26,779	(ND)	(ND)
B-923	12	45.7	5.33	2.39	2.2	163	Shear	13,477	(ND)	(ND)
B-923	16	65.7	5.35	2.39	2.2	164	Cone	21,069	7,150,000	0.29
B-924	1	21.7	5.33	2.39	2.2	162	Shear	10,588	(ND) 3	(ND)
B-924	3	30.2	5.35	2.39	2.2	163	C&S	15,110	(ND)	(ND)
B-924	6	44.0	5.33	2.39	2.2	174	Shear	6,384	2,620,000	* (4)
B-924	12	75.1	5.33	2.40	2.2	179	C&S	5,681	(ND)	(ND)
B-927	2	43.0	5.35	2.39	2.2	163	C&S	19,288	(ND)	(ND)
B-927	6	51.6	5.35	2.39	2.2	163	C&S	27,239	6,550,000	0.49
B-927	13	74.9	5.33	2.39	2.2	164	Cone	30,297	(ND)	(ND)
B-927	18	96.3	5.35	2.39	2.2	164	C&S	28,266	(ND)	(ND)
B-928	2	52.6	5.33	2.39	2.2	153	Shear	1,318	(ND)	(ND)

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Table 2.5-211 Results of Unconfined Compression Tests on Rock

Boring No.	Run Number	Sample Top Depth (ft)	Sample Length (L) (Inches)	Sample Diameter (D) (inches)	L/D Ratio	Unit Weight (pcf)	Type of Break (1)	Unconfined Compressive Strength (psi)(2)	Young's Modulus (psi)	Poisson's Ratio
B-928	6	74.7	5.35	2.39	2.2	162	Cone	20,333	5,070,000	0.35
B-933	3	50.5	5.33	2.39	2.2	163	Cone	19,395	(ND)	(ND)
B-933	7	66.6	5.34	2.38	2.2	162	Columnar	15,764	8,600,000	* (4)
B-933	11	90.1	5.32	2.39	2.2	164	Cone	30,993	(ND)	(ND)
B-948	6	56.8	5.28	2.39	2.2	162	C&S	17,089	(ND)	(ND)
B-948	10	76.1	5.25	2.40	2.2	167	C&S	22,435	(ND)	(ND)

(1) Type of Breaks: Columnar; Cone (C); Shear (S); Cone & Shear (C&S)

(2) Unconfined compressive strength corrected for L/D Ratio
Compressive strength testing was performed in general accordance with ASTM D7012-04.

(3) (ND) indicates that information was not determined

(4) Value of Poisson's ratio is greater than 0.5 which indicates inelastic behavior probably due to presence of fractures or discontinuities affecting lateral strain.

NAPS COL 2.0-29-A

Table 2.5-212 Engineering Properties for Soil and Bedrock

Stratum	Structural Fill	Zone IIA	Zone IIB	Zone III	Zone III-IV	Zone IV
Description	**Gravelly materials derived from crushing rock material**	**Saprolite – core stone less than 10% of volume of overall mass**	**Saprolite – core stone 10% to 50% of volume of overall mass**	**Weathered rock – core stone more than 50% of volume of overall mass**	**Moderately weathered to slightly weathered rock**	**Parent rock – slightly weathered to fresh rock**
USCS symbol	GW	SM, SC	SM	-	-	-
Total unit weight, g (pcf)	130	125	130	150	163	164
Fines Content (%)	0-5	25	20	-	-	-
Natural water content, w_N (%)	-	19	14	-	-	-
Atterberg limits						
Liquid limit, LL	-	-	-	-	-	-
Plastic limit, PL	-	-	-	-	-	-
Plasticity index, PI	-	-	-	-	-	-
Measured SPT N-value (blows/ft)	-	15	75	Ref	-	-
Adjusted SPT N_{60}-value (blows/ft)	50	20	100	Ref	-	-
Undrained properties						
Undrained shear strength, s_u (ksf)	-	-	-	-	-	-
Unconfined compressive strength, q_u(ksi)	-	-	-	1.0	9.0	17.0
Drained properties						
Effective cohesion, c' (ksf)	0	0.125	0	-	-	-
Effective friction angle, ' (degrees)	40	33	40	-	-	-

NAPS COL 2.0-29-A

Table 2.5-212 Engineering Properties for Soil and Bedrock

Stratum	Structural Fill	Zone IIA	Zone IIB	Zone III	Zone III-IV	Zone IV
Description	**Gravelly materials derived from crushing rock material**	**Saprolite – core stone less than 10% of volume of overall mass**	**Saprolite – core stone 10% to 50% of volume of overall mass**	**Weathered rock – core stone more than 50% of volume of overall mass**	**Moderately weathered to slightly weathered rock**	**Parent rock – slightly weathered to fresh rock**
Shear wave velocity, V_s (ft/sec)	1,100	850	1,600	3,000	4,500	9,000
Compression wave velocity, V_p (ft/sec)	2,400	1,800	3,500	7,300	9,000	16,000
Poisson's ratio, u (high strain)	0.3	0.35	0.3	0.4	0.33	0.27
Poisson's ratio, u (low strain)	0.37	0.35	0.37	0.4	0.33	0.27
Elastic modulus (high strain), E_h	1,800 ksf	720 ksf	3,600 ksf	400 ksi	1,900 ksi	7,250 ksi
Elastic modulus (low strain), E_l	13,000 ksf	7,500 ksf	28,000 ksf	800 ksi	1,900 ksi	7,250 ksi
Shear modulus (high strain), G_h	700 ksf	270 ksf	1,400 ksf	150 ksi	700 ksi	2,900 ksi
Shear modulus (low strain), G_l	5,000 ksf	2,800 ksf	10,000 ksf	300 ksi	700 ksi	2,900 ksi
Consolidation characteristics						
Compression ratio, CR				-	-	-
Recompression ratio, RR				-	-	-
Coefficient of subgrade reaction, k_1 (kcf)	2,000	260	2,000	-	-	-
Coefficient of sliding	0.55	0.35	0.45	0.6	0.65	0.7

NAPS COL 2.0-29-A

Table 2.5-212 Engineering Properties for Soil and Bedrock

Stratum	Structural Fill	Zone IIA	Zone IIB	Zone III	Zone III-IV	Zone IV
Description	**Gravelly materials derived from crushing rock material**	**Saprolite – core stone less than 10% of volume of overall mass**	**Saprolite – core stone 10% to 50% of volume of overall mass**	**Weathered rock – core stone more than 50% of volume of overall mass**	**Moderately weathered to slightly weathered rock**	**Parent rock – slightly weathered to fresh rock**
Static earth pressure coefficients						
Active, K_a	0.22	0.30	0.22	-	-	-
Passive, K_p	4.60	3.40	4.60	-	-	-
At-rest, K_0	0.36	0.50	0.36	-	-	-
Optimum moisture content, w_{opt} (%)		14		-	-	-
Maximum dry unit weight, g_{max} (pcf)		116		-	-	-
Rock Quality Designation, RQD (%)	-	-	-	20	65	95

NAPS ESP COL 2.5-6

Table 2.5-213 Summary of Major Structures

Structure	Seismic Category	Approximate Dimensions (ft)	Bottom of Foundation Elevation[1] (ft)	Embed-ment Depth (ft)	Design Load (ksf)	
					Static	Dynam-ic
Reactor/Fuel Building	I	161 x 230	223.9	65.6	14.6	112.8
Control Building	I	78 x 99	240.6	48.9	6.1	50.2
Fire Water Service Complex	I	66 x 171	281.8	7.7	3.45	14.0
Turbine Building	NS	194 x 377	263.6	25.9	6	—
Radwaste Building	NS[2]	108 x 213	237.5	52.0	6	—
Service Building	II	111 x 163	274.1	15.4	4	—
Ancillary Diesel Building	II	61 x 71	286.2	3.3	4	—

Note: (1) The bottom of foundation is derived from the finished ground level grade at Elevation 289.5 ft.

(2) The Radwaste Building is seismically designed. See DCD Table 2.0-1, Note 1.

NAPS ESP COL 2.5-6

Table 2.5-214 Allowable Static Bearing Capacities of Rock

Rock Type	Unconfined Compressive Strength, q_u (ksi)	$q_a = 0.2\ q_u$ (ksf)	Recommended q_a (ksf)
Zone III	1	29	20
Zone III-IV	9	259	80
Zone IV	17	490	160

NAPS ESP COL 2.5-6

Table 2.5-215 Summary of Allowable Bearing Capacities for the Major Structures

	Calculated Allowable Bearing Capacity, q_a (ksf)						Minimum q_a (ksf)	
Structure	Structural Fill	Concrete Fill	Zone IIB	Zone III	Zone III-IV	Zone IV	Static	Dy-namic
Reactor/Fuel Building	-	214	-	-	80	160	80	214
Control Building	-	214	-	20	80	160	50	144
Fire Water Service Complex	83.4	-	-	20	80	160	20	29
Turbine Building	242.5	-	242.5	20	80	160	20	29
Radwaste Building	214.1	-	-	20	80	160	20	29
Service Building	134.9	-	134.9	20	80	160	20	29
Ancillary Diesel Building	—	—	57.5	20	80	160	20	29

NAPS ESP COL 2.5-6

Table 2.5-216 Estimated Settlements of the Major Structures

Structure	Applied Load (ksf)	Settlement (in.)			
		Center	Edge	Average(1)	Corner
Reactor/Fuel Building	14.6	0.12	0.08	0.10	0.05
Control Building	6.1	0.04	0.03	0.035	0.02
Fire Water Service Complex	3.45(2) 2.30(3)	0.94 0.62	0.51 0.34	0.73 0.48	0.26 0.17
Turbine Building	6	2.24	1.14	1.69	0.58
Radwaste Building	6	0.75	0.38	0.57	0.19
Service Building	4	1.56	0.83	1.20	0.43
Ancillary Diesel Building	4	0.14	0.07	0.11	0.04

Notes: (1) Average is average of center and edge settlements
(2) Applied load including weight of basemat
(3) Applied load excluding weight of basemat

NAPS ESP COL 2.5-10

Table 2.5-217 Maximum Acceleration Results

Depth (ft)	Low Frequency Max. Acc. (g)	High Frequency Max. Acc. (g)
0.0	0.2964	0.5531
2.5	0.2693	0.5237
5.0	0.2338	0.4691
7.5	0.2200	0.4461
10.0	0.2099	0.4356
12.5	0.2065	0.4444
15.0	0.2065	0.4692
17.5	0.2079	0.4761
20.0	0.2088	0.4841
22.5	0.2112	0.4831
25.0	0.2200	0.4975
27.5	0.2266	0.5042
30.0	0.2291	0.5180
32.5	0.2279	0.5366
35.0	0.2273	0.5510
37.5	0.2265	0.5467
40.0	0.2219	0.5367
42.5	0.2164	0.5275
45.0	0.2091	0.5115
50.0	0.1881	0.4395
55.0	0.1794	0.4085

NAPS COL 2.0-30-A

Table 2.5-218 Water Level Measurements for Well OW-947

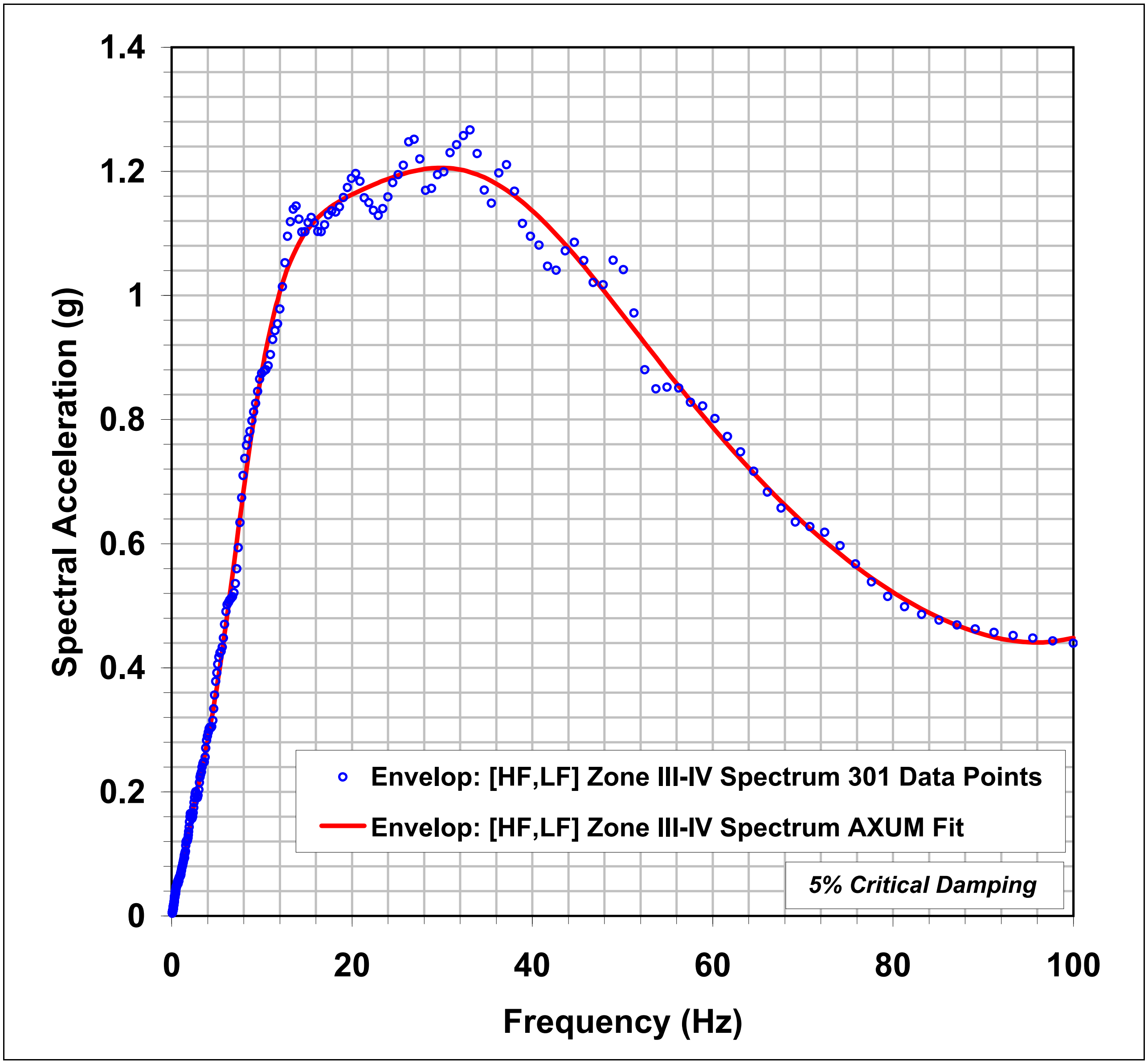
Date	Groundwater Elevation, Ft
11/29/2006	297.61
2/28/2007	297.81
5/30/2007	297.92
8/29/2007	296.00

NAPS COL 2.0-30-A

Table 2.5-219 Grain-Size Test Results for Boring B-947

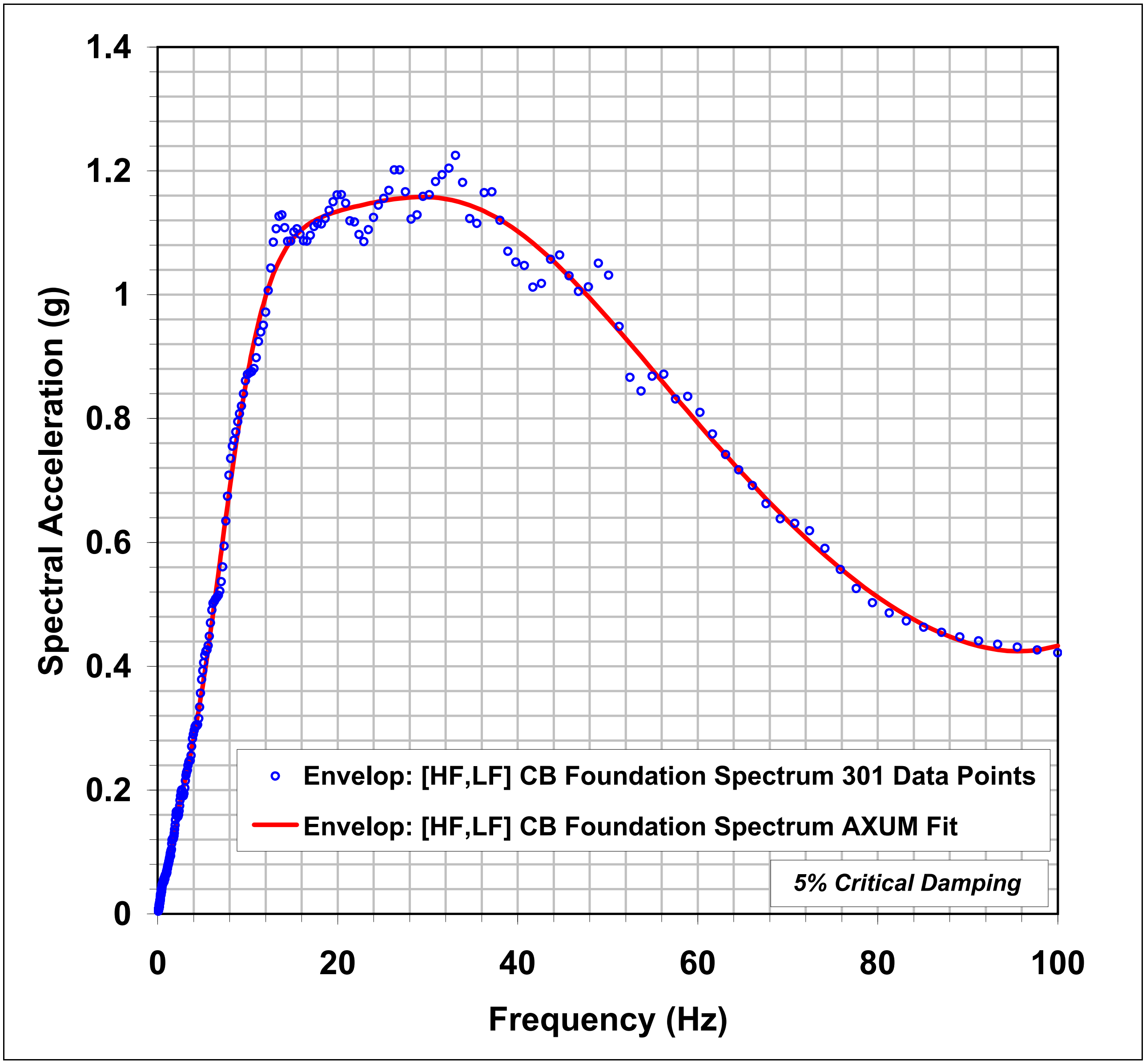
Sample No.	Depth (Ft)	Gravel (%)	Sand (%)	Fines (%)	Silt (%)	Clay (%)
B-947-3	4.5–6.0	0.0	38.3	61.7	23.5	38.2
B-947-4	8.5–10.0	0.0	60.0	40.0	-	-
B-947-5	9.5–11.0	1.6	55.9	42.5	21.1	21.4
B-947-6	13.5–15.0	0.0	30.5	69.5	-	-
B-947-7	17.2–18.7	0.0	75.8	24.2	-	-
B-947-8	22.2–23.7	0.6	79.4	20.0	10.7	9.3
B-947-9	28.7–30.2	0.0	66.6	33.4	-	-
B-947-10	33.7–35.2	0.0	81.3	18.7	-	-
B-947-11	38.7–40.2	0.0	85.8	14.2	-	-
B-947-12	42.2–43.7	0.0	79.7	20.3	13.4	6.9

NAPS COL 2.0-27-A **Figure 2.5-201 Plot of the 301-Point Response Spectrum Processed from SHAKE Output and the Smooth Fitting Function for the Control Point, Zone III-IV, Top of Competent Rock (Elevation 83.2 m (273 ft))**



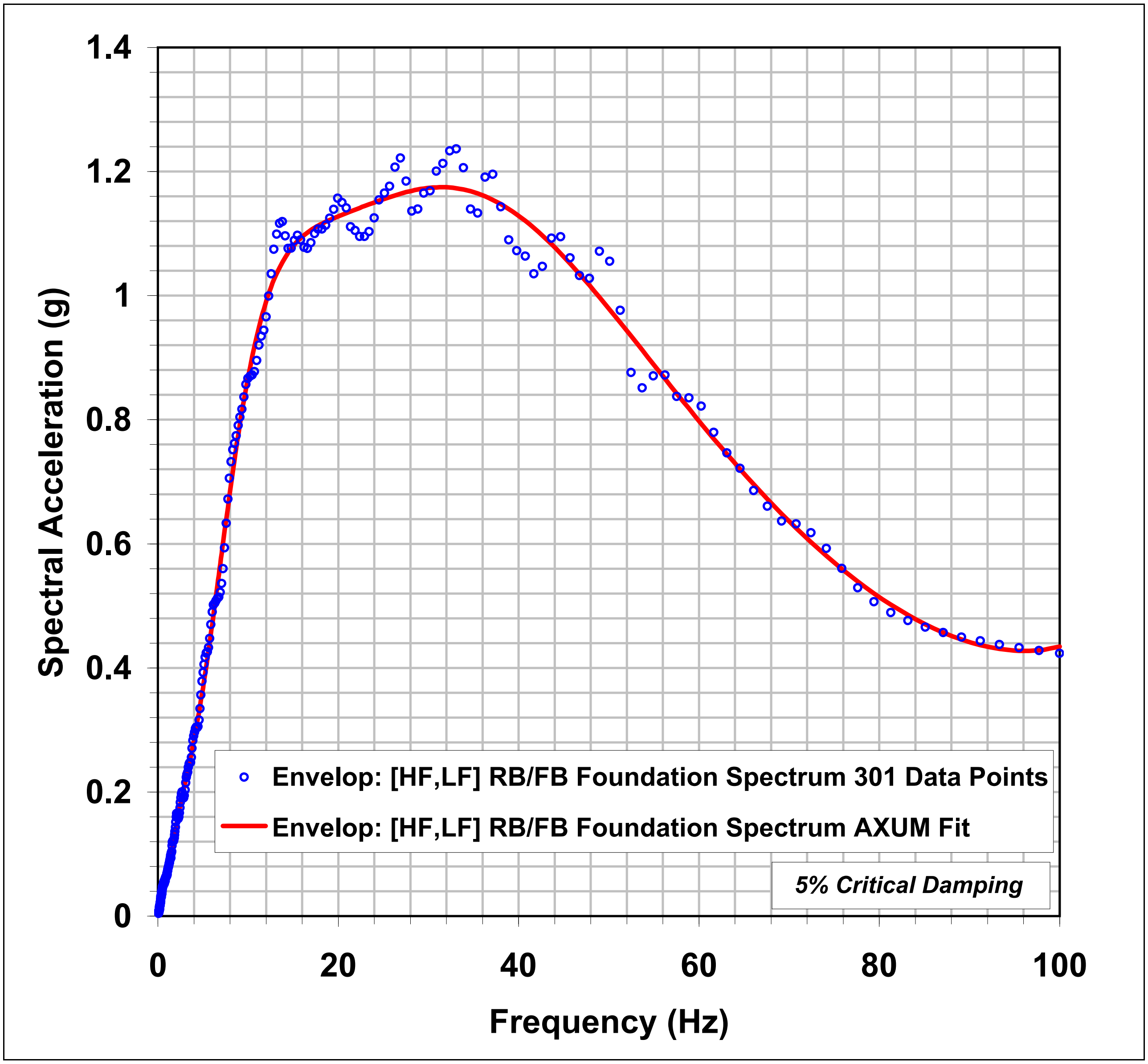
NAPS COL 2.0-27-A

Figure 2.5-202 Plot of the 301-Point Response Spectrum Processed from SHAKE Output and the Smooth Fitting Function for the Base of CB Foundation (Elevation 73.5 m (241 ft))



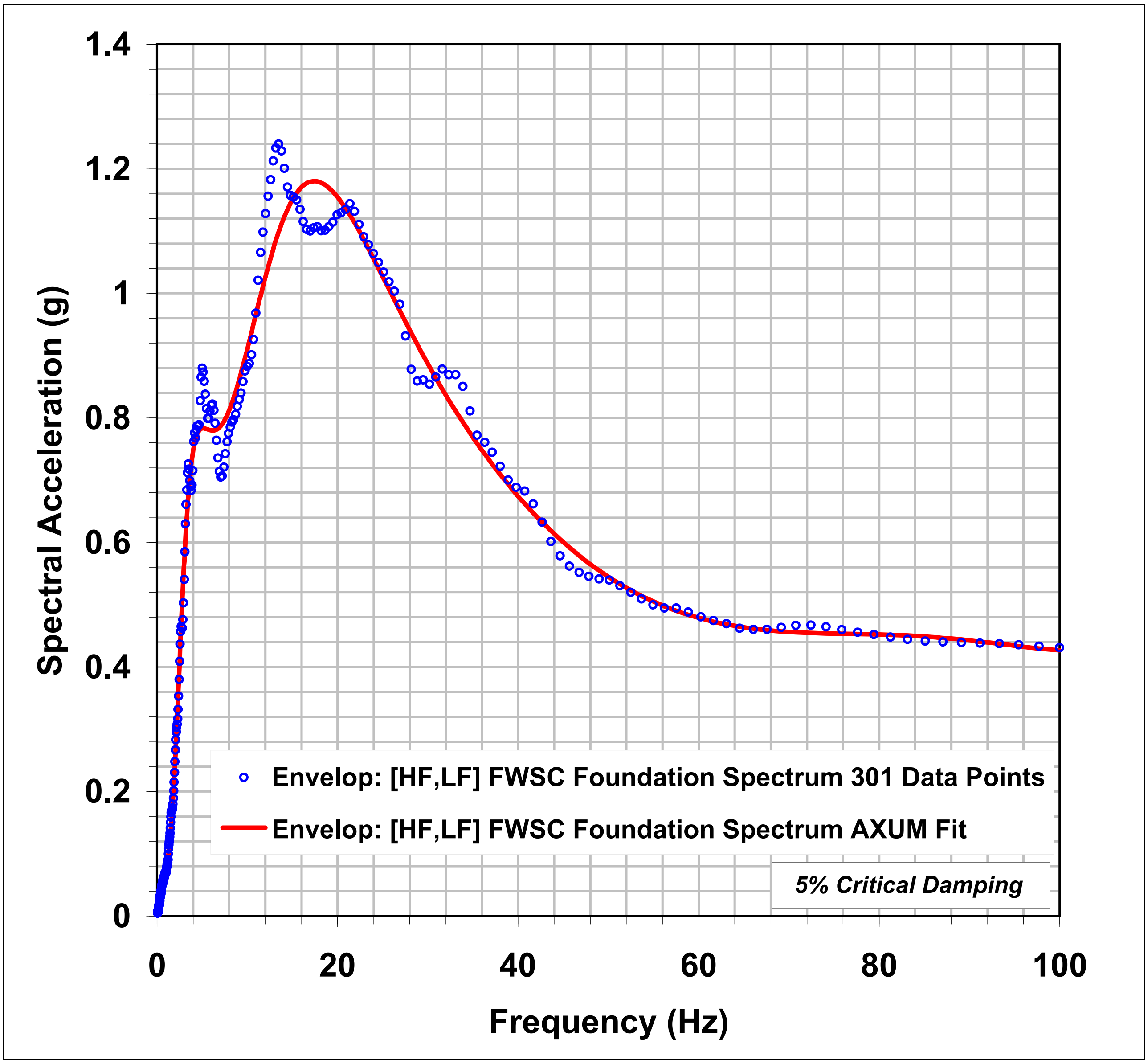
NAPS COL 2.0-27-A

Figure 2.5-203 Plot of the 301-Point Response Spectrum Processed from SHAKE Output and the Smooth Fitting Function for the Base of the RB/FB Foundation (Elevation 68.3 m (224 ft))

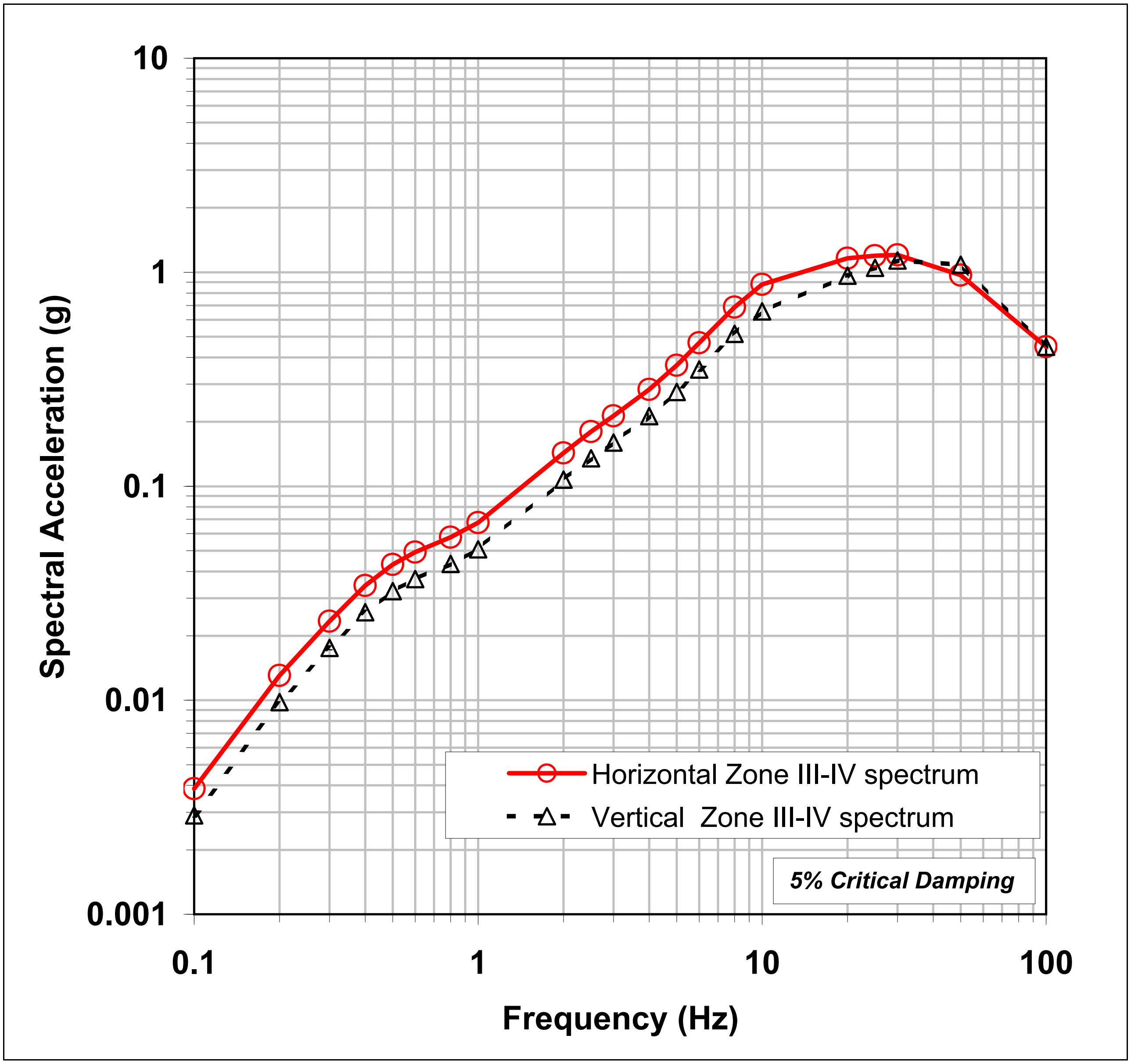


NAPS COL 2.0-27-A

Figure 2.5-204 Plot of the 301-Point Response Spectrum Processed from SHAKE Output and the Smooth Fitting Function for the Base of the FWSC Foundation (Elevation 86.0 m (282 ft))

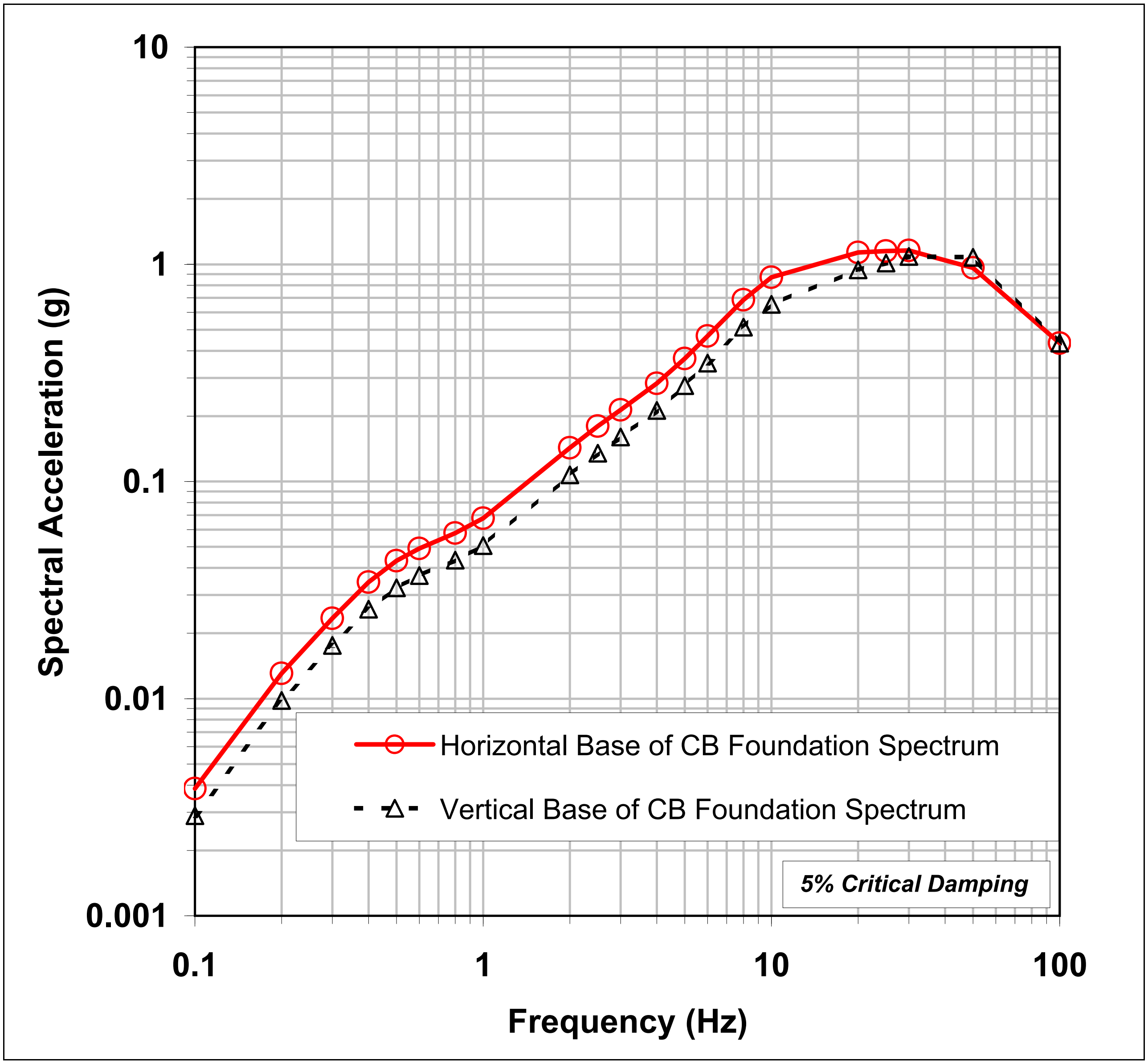


NAPS COL 2.0-27-A **Figure 2.5-205 Selected Horizontal and Vertical Control Point SSE Response Spectra at the Top of Zone III-IV Material, Top of Competent Rock (Elevation 83.2 m (273 ft))**



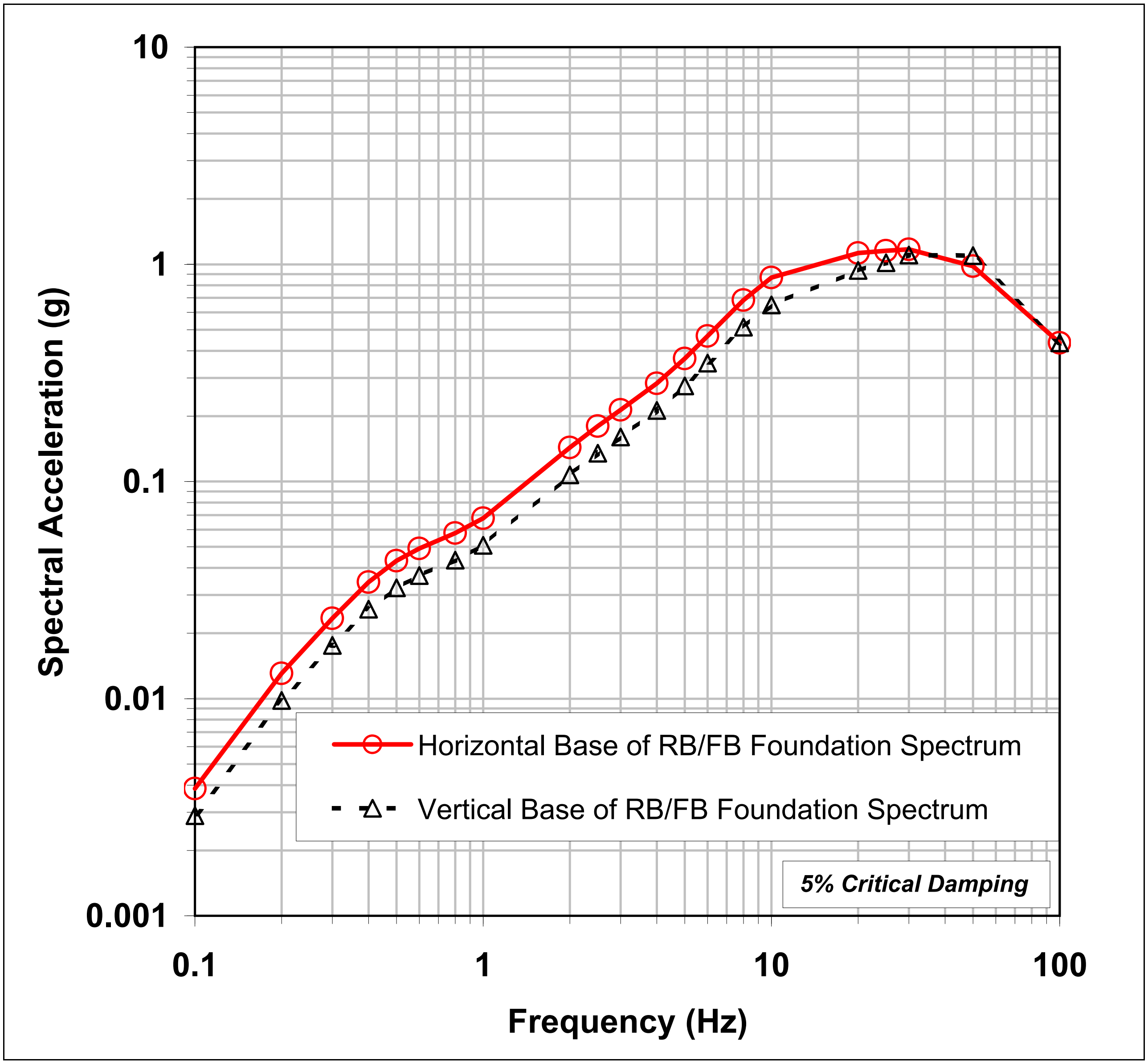
NAPS COL 2.0-27-A

Figure 2.5-206 Selected Horizontal and Vertical SSE Response Spectra at the Base of the CB Foundation (Elevation 73.5 m (241 ft))

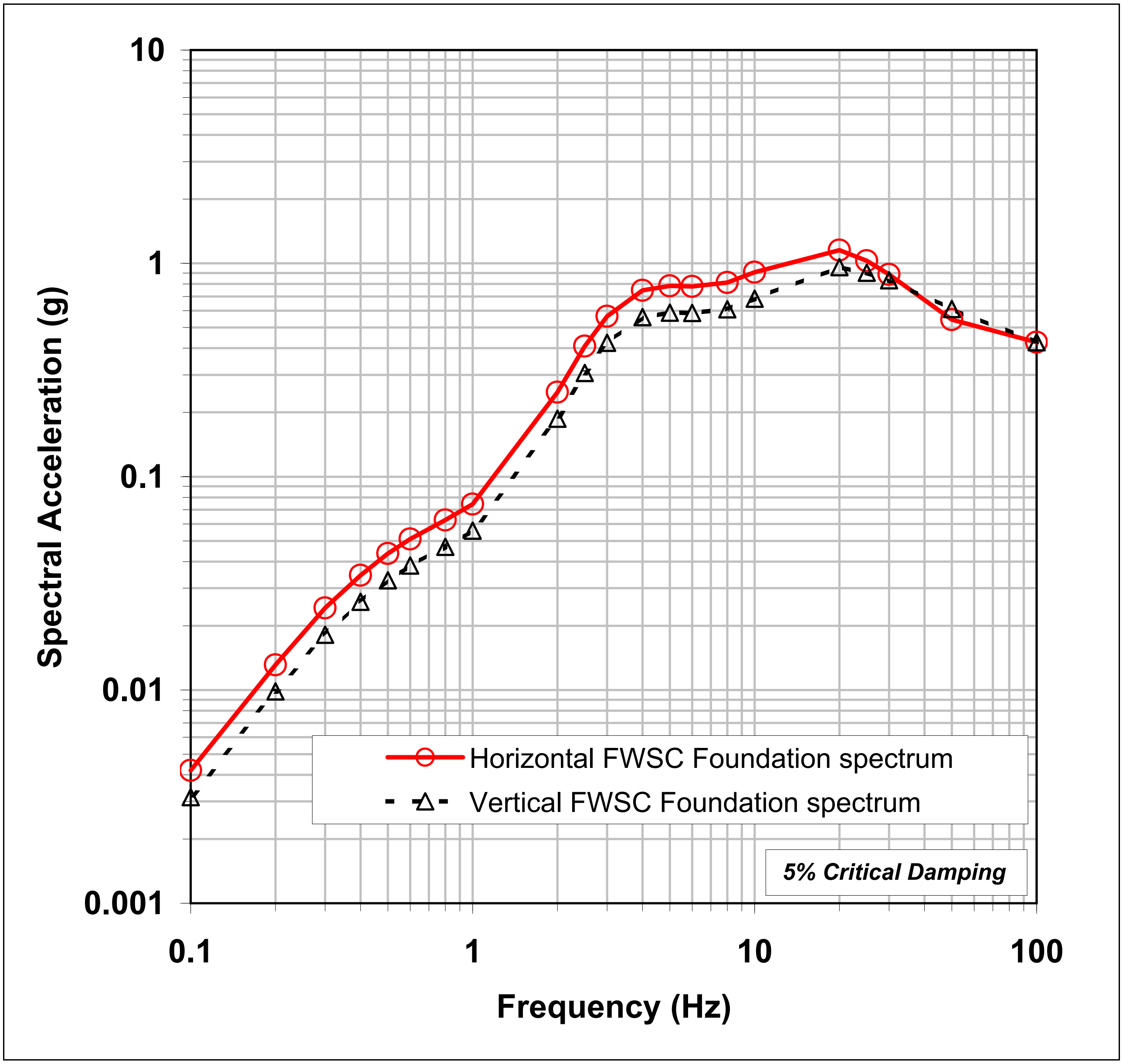


NAPS COL 2.0-27-A

Figure 2.5-207 Selected Horizontal and Vertical SSE Response Spectra at the Base of the RB/FB Foundation (Elevation 68.3 m (224 ft))



NAPS COL 2.0-27-A **Figure 2.5-208 Selected Horizontal and Vertical SSE Response Spectra at the Base of the FWSC Foundation (Elevation 86.0 m (282 ft))**



NAPS COL 2.0-29-A **Figure 2.5-209 Contours of Top of Zone IV**

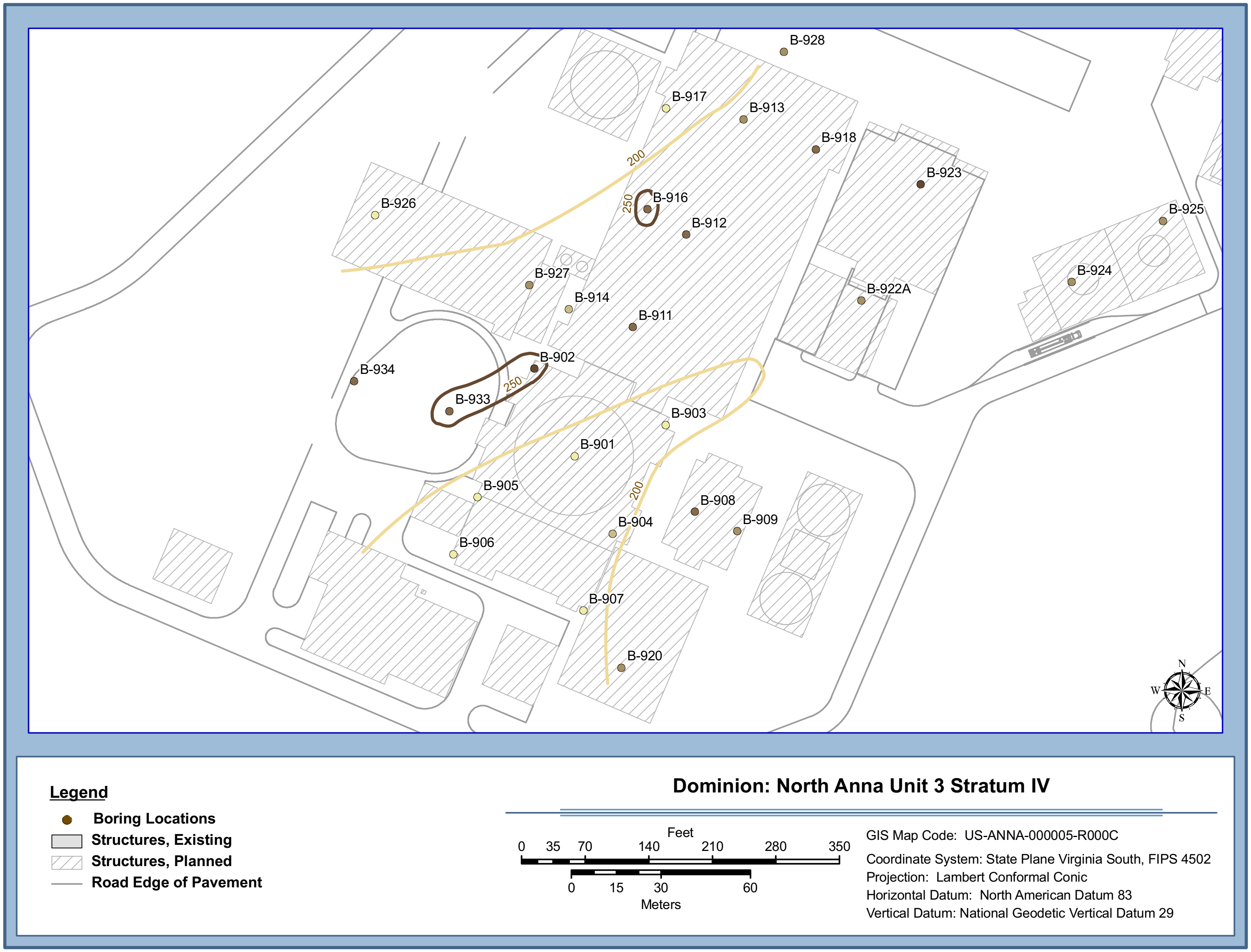


Figure 2.5-210 Contours of Top of Zone III-IV

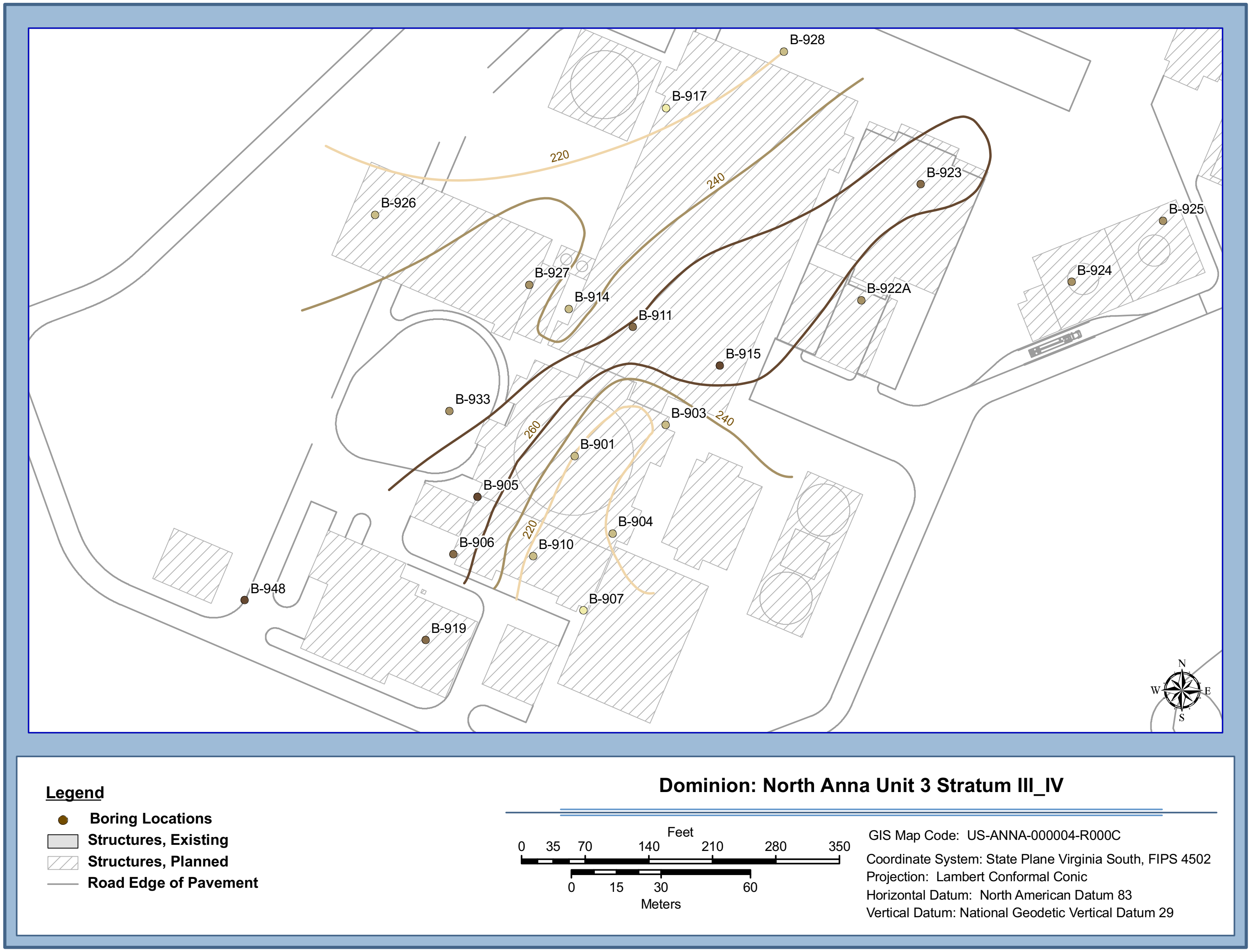
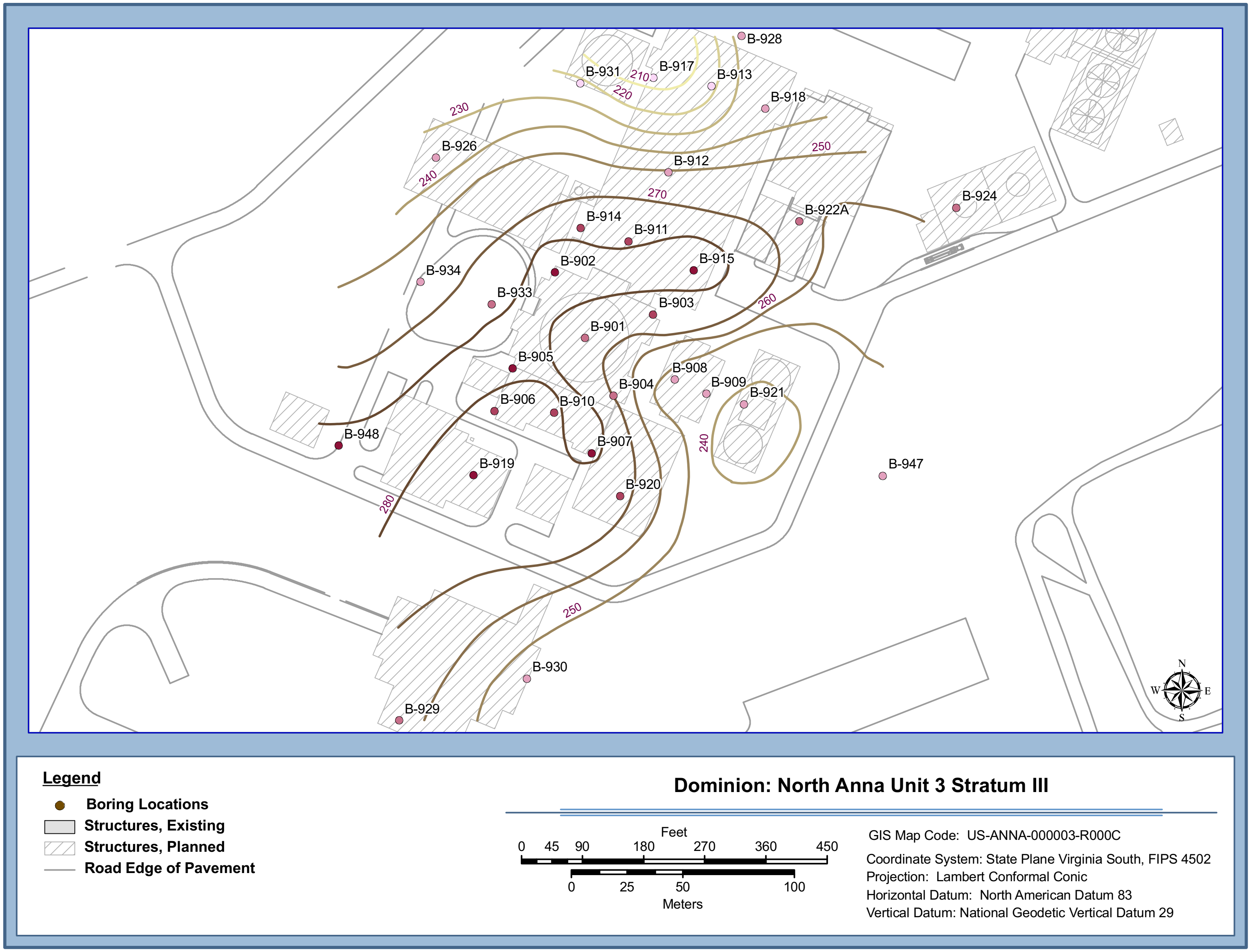
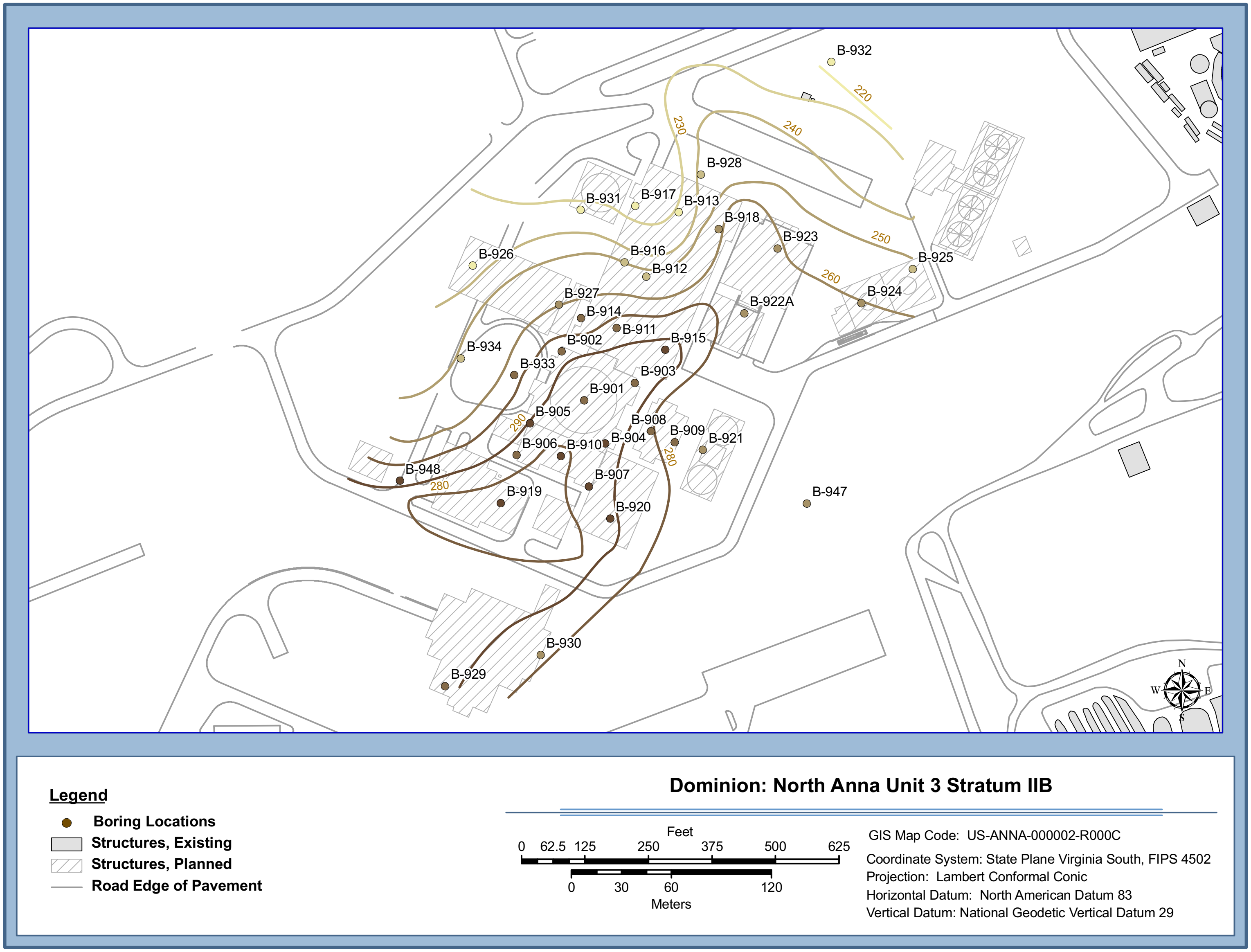


Figure 2.5-211 Contours of Top of Zone III

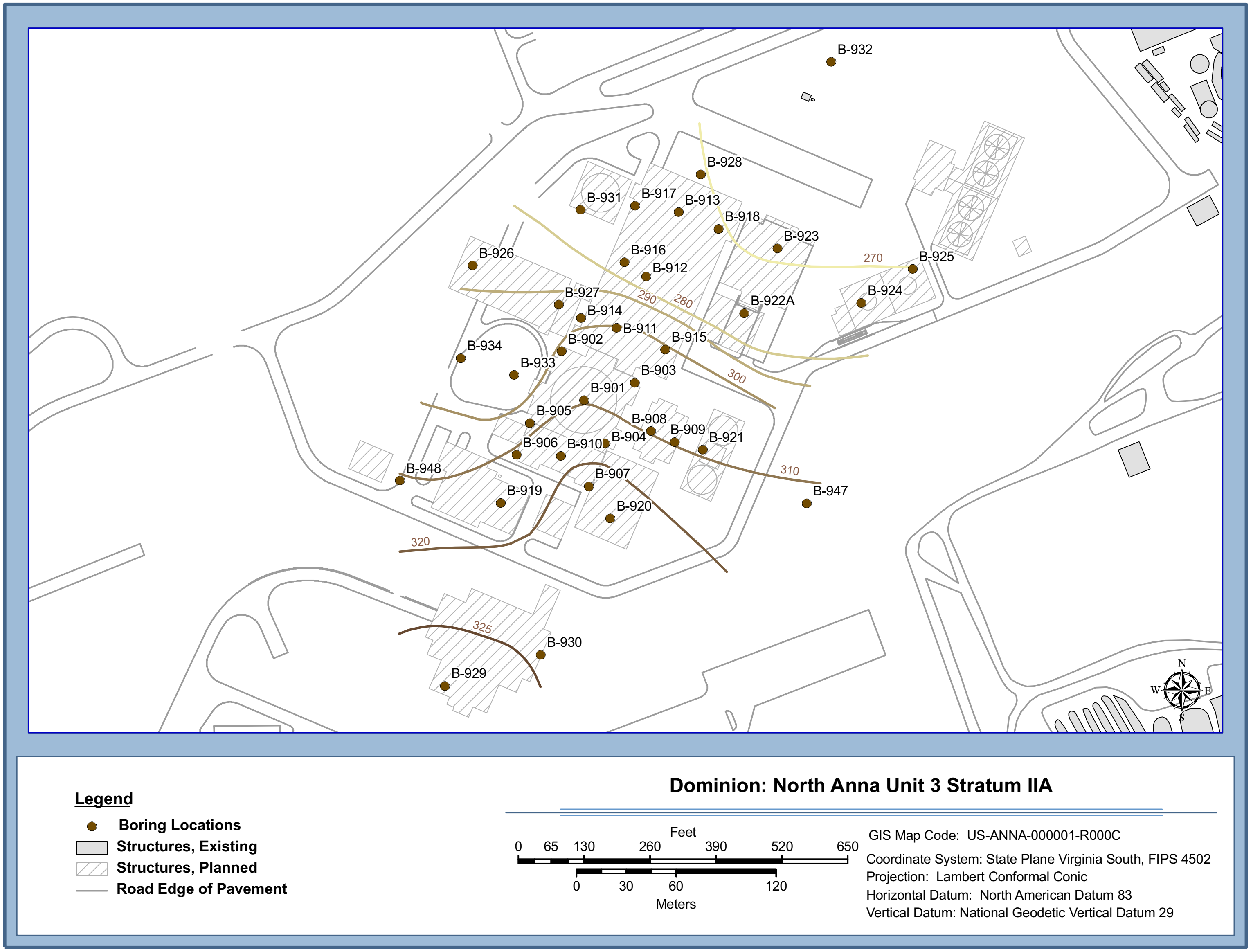


NAPS COL 2.0-29-A **Figure 2.5-212 Contours of Top of Zone IIB**



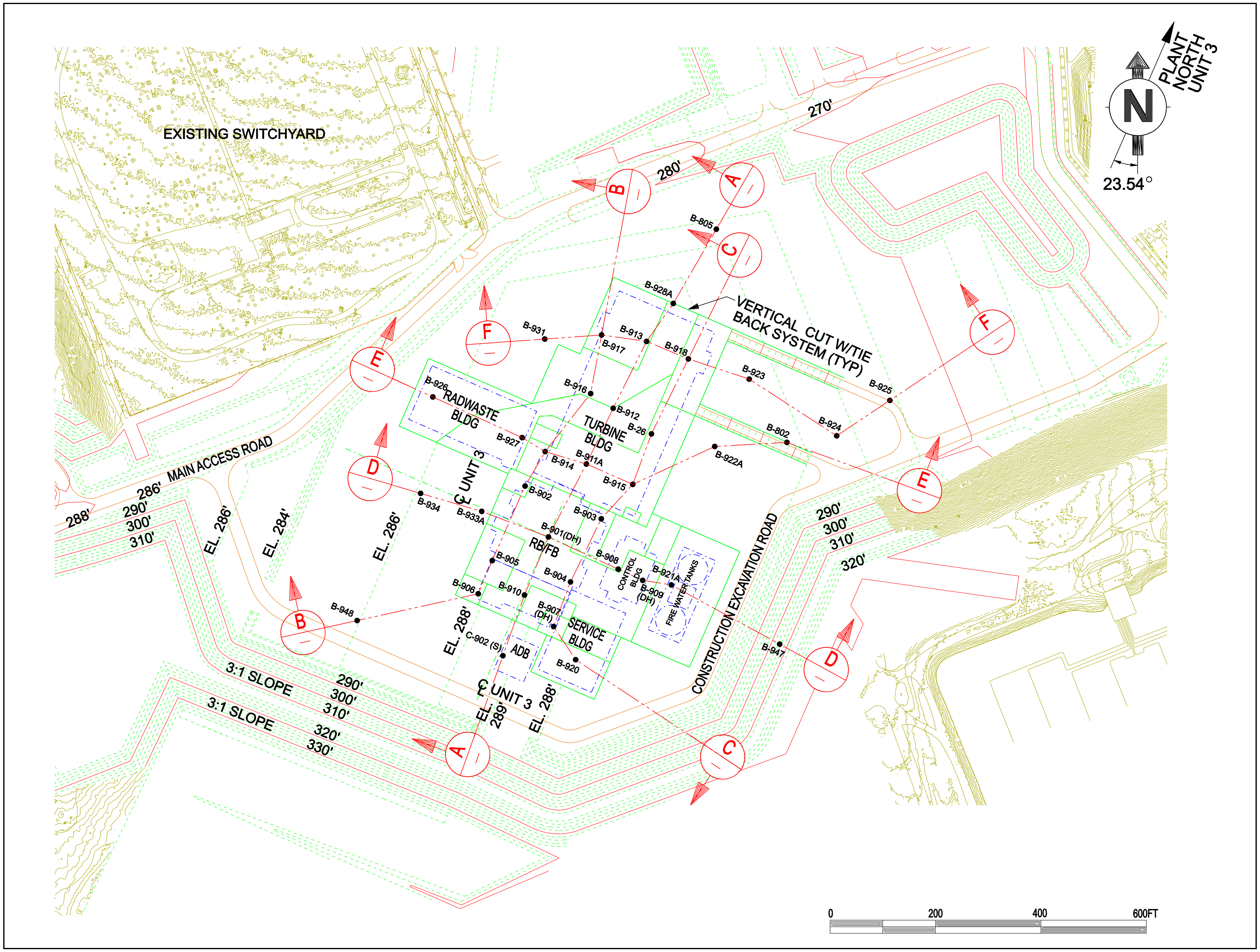
NAPS COL 2.0-29-A

Figure 2.5-213 Contours of Top of Zone IIA



NAPS COL 2.0-29-A

Figure 2.5-214 Plan Locations of Profiles



NAPS COL 2.0-29-A **Figure 2.5-215 Subsurface Profile A-A'**

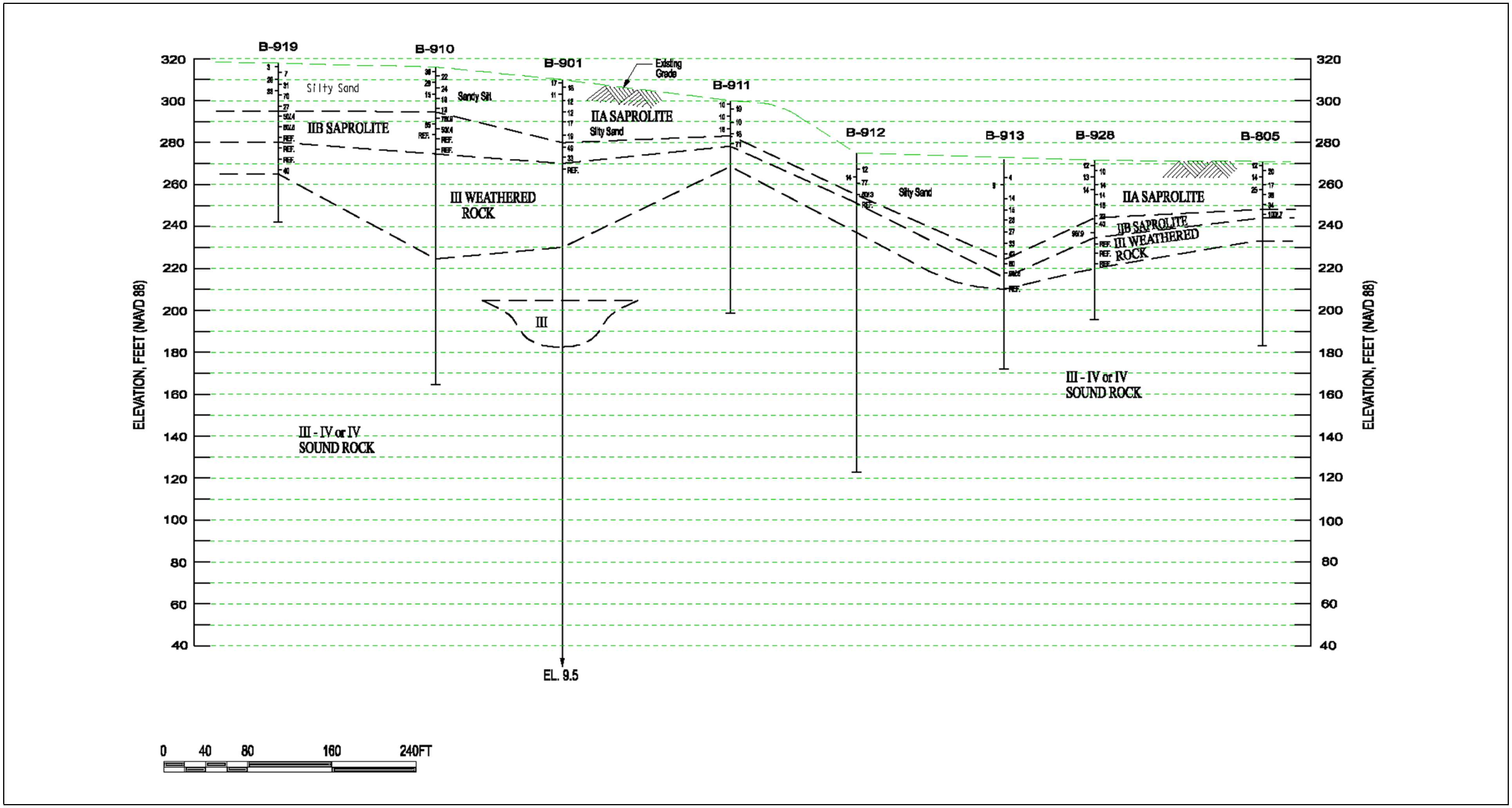
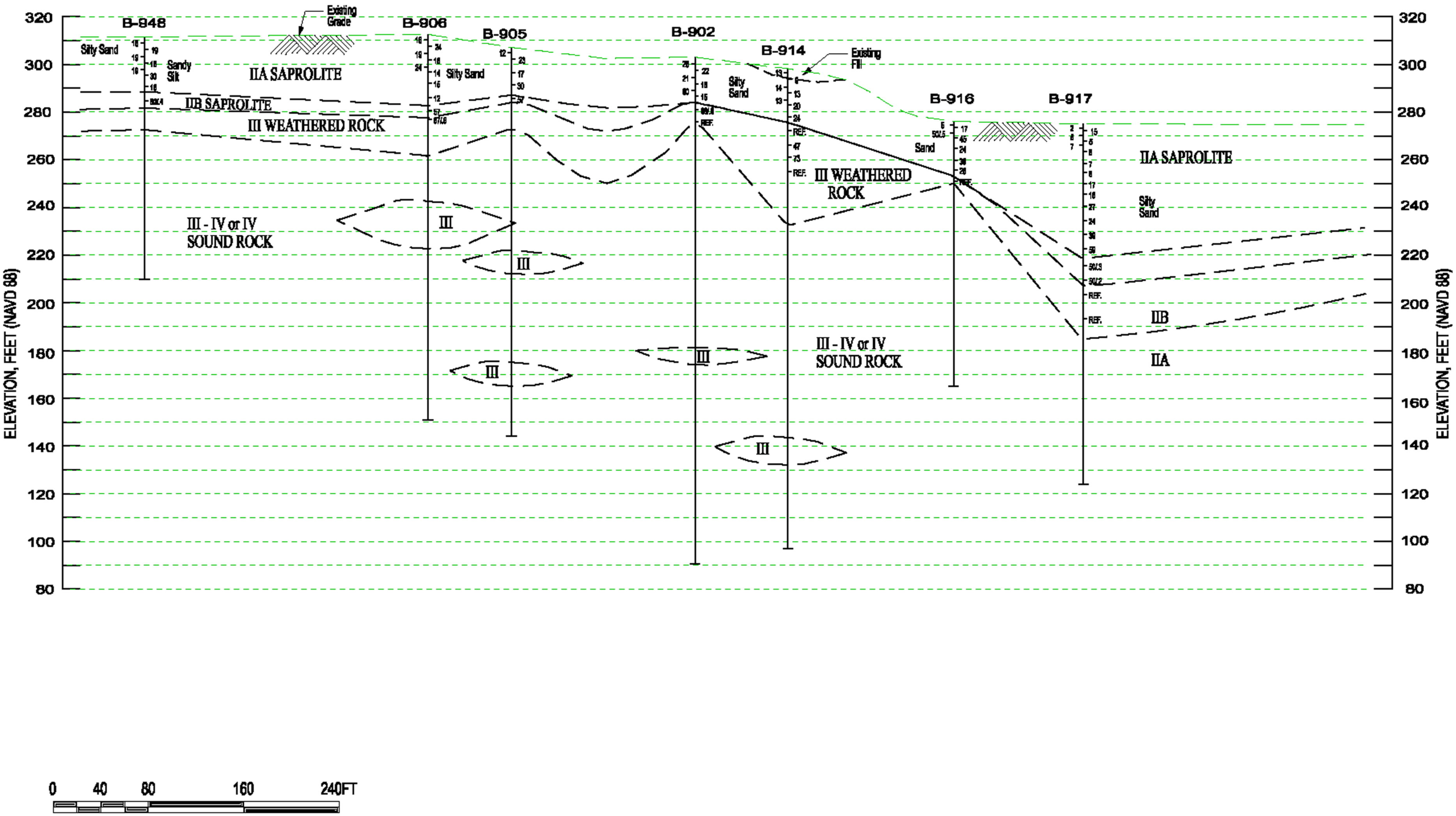


Figure 2.5-216 Subsurface Profile B-B'



NAPS COL 2.0-29-A **Figure 2.5-217 Subsurface Profile C-C'**

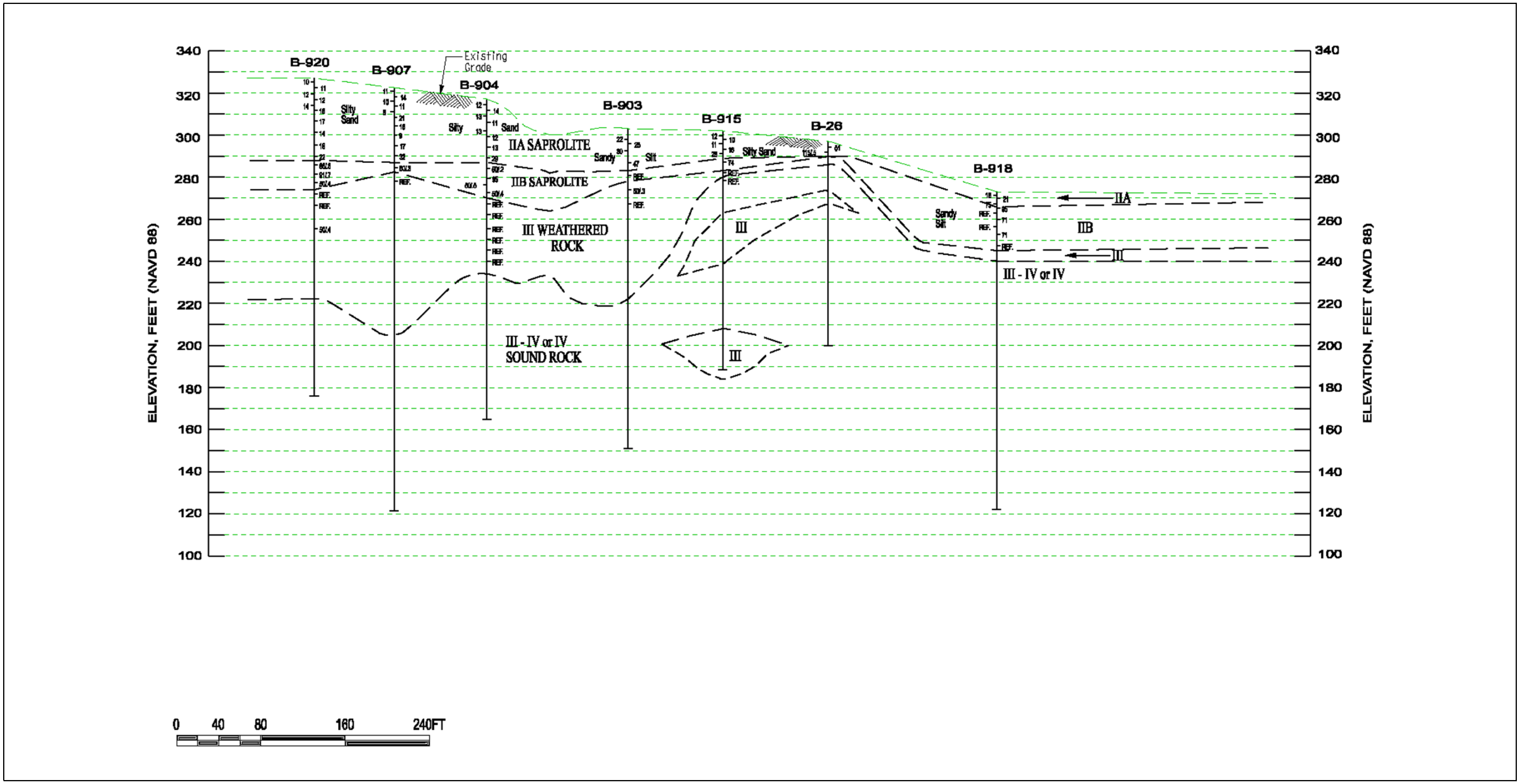
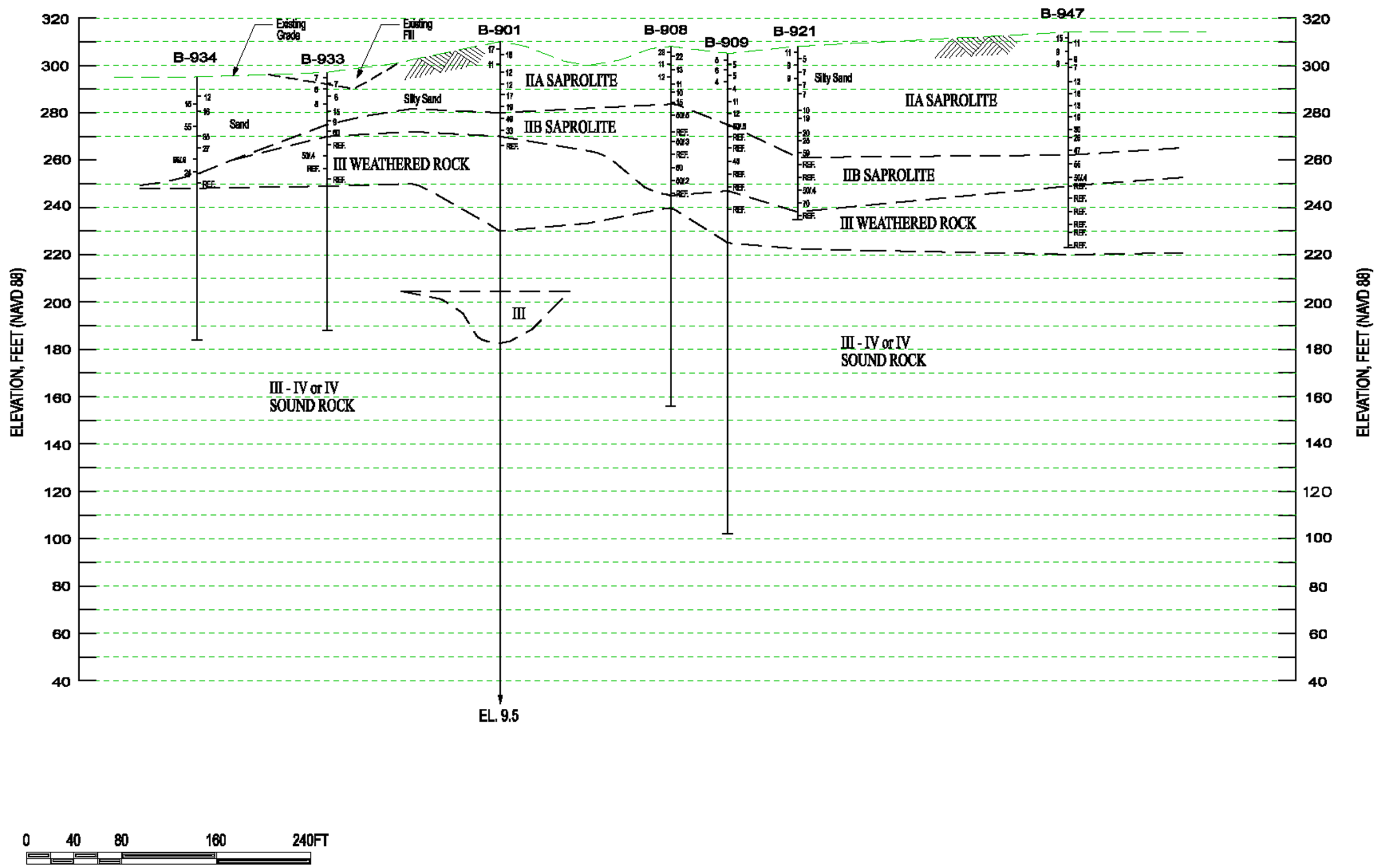


Figure 2.5-218 Subsurface Profile D-D'



NAPS COL 2.0-29-A **Figure 2.5-219 Subsurface Profile E-E'**

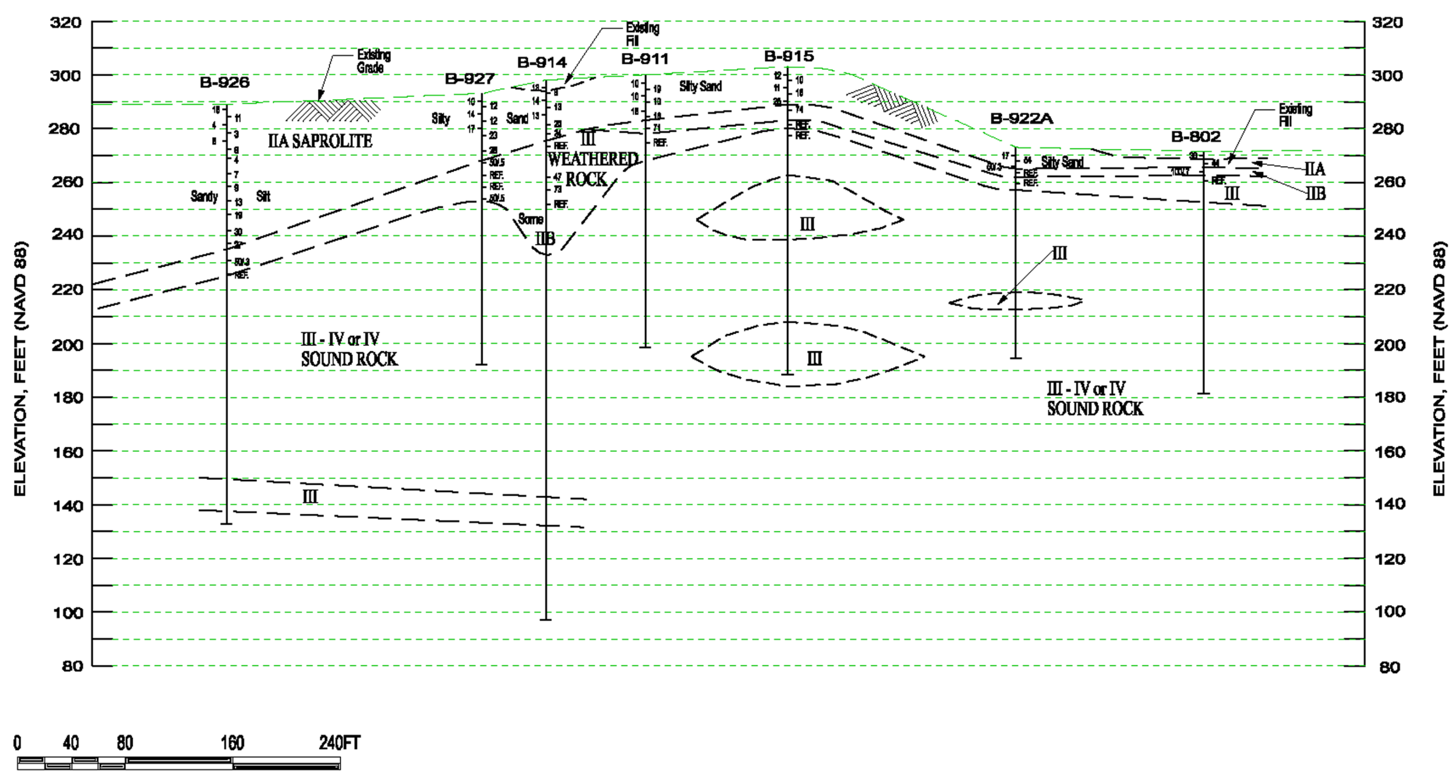


Figure 2.5-220 Subsurface Profile F-F'

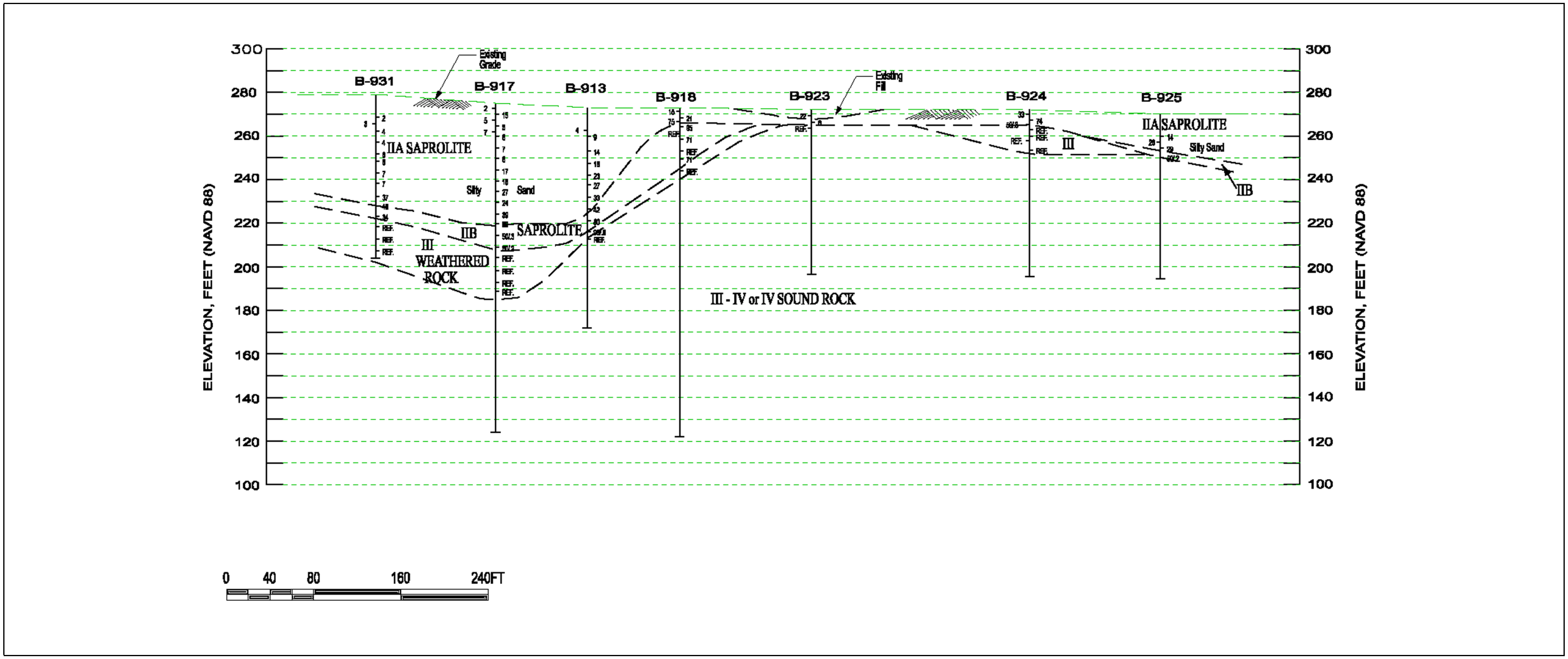


Figure 2.5-221 Unit 3 Boring Locations – Power Block

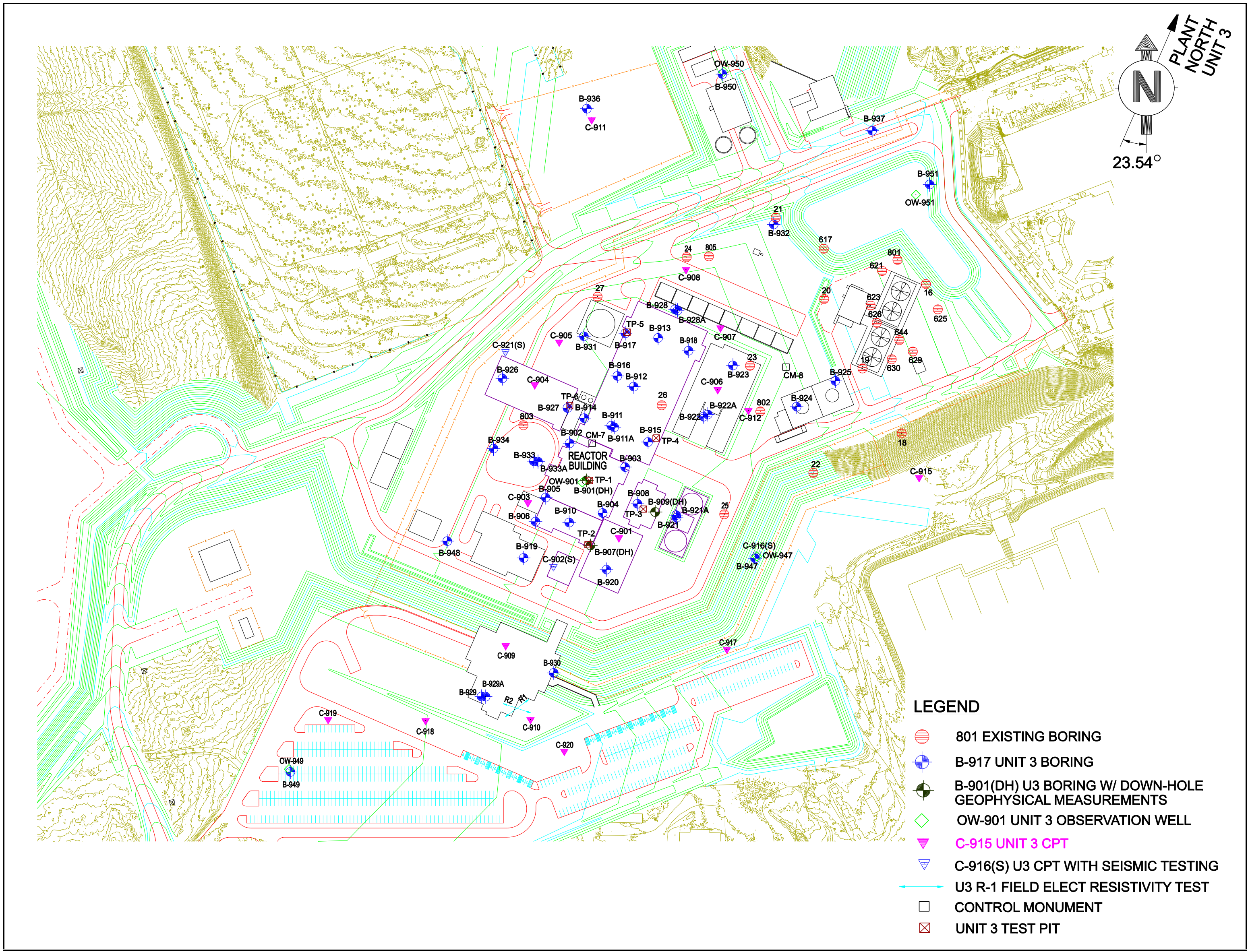
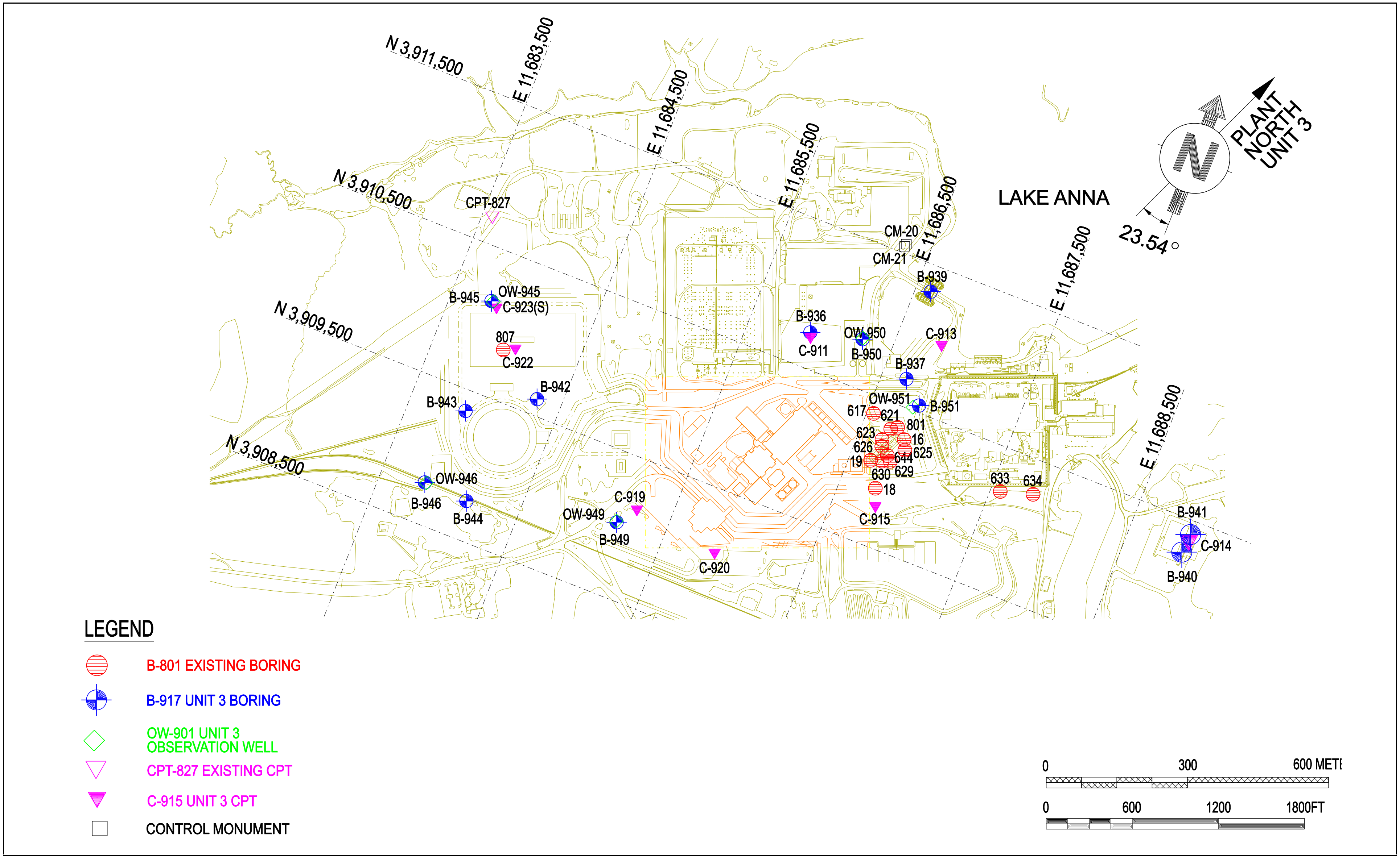
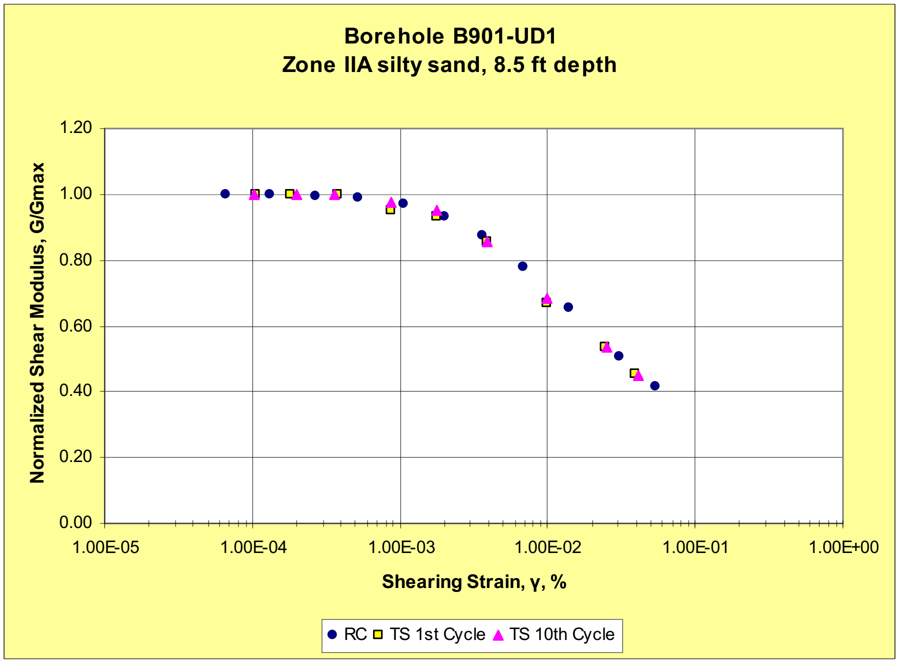


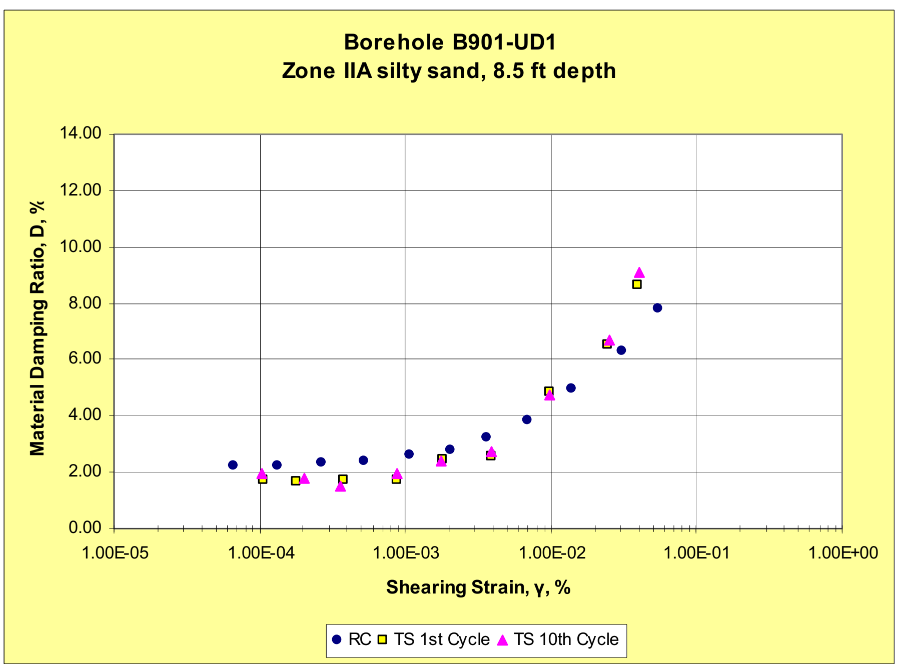
Figure 2.5-222 Unit 3 Boring Locations – Site



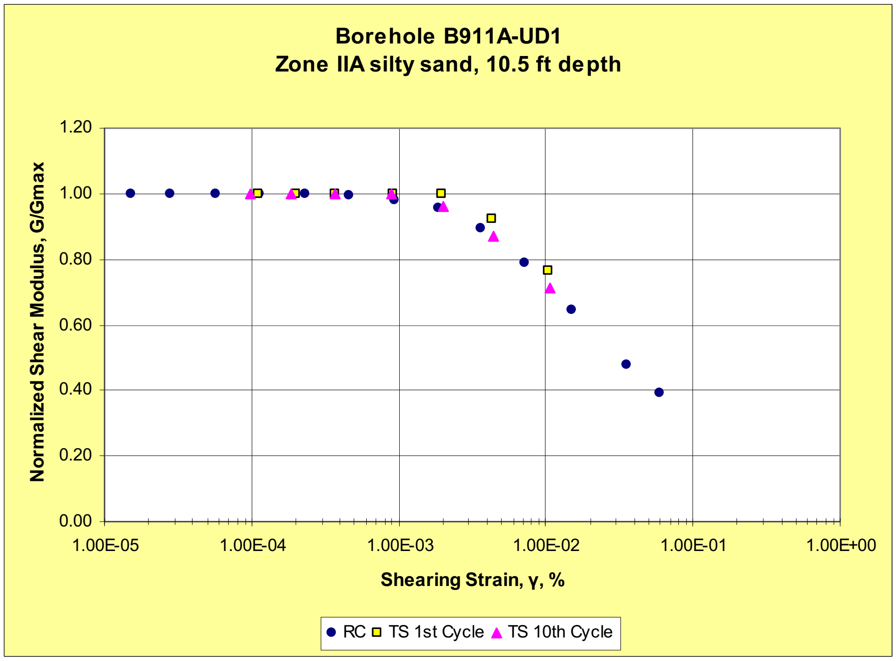
NAPS COL 2.0-29-A

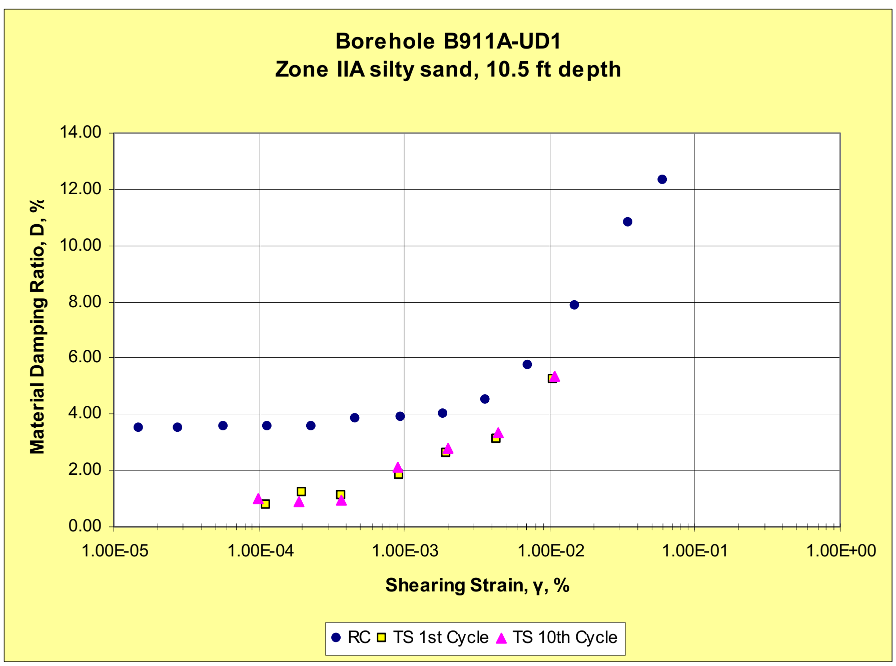
Figure 2.5-223 RCTS Test Results (Sheet 1 of 3) G/G_{max} and D vs. Strain, B-901 UD-1: 4.3 psi Confining Pressure



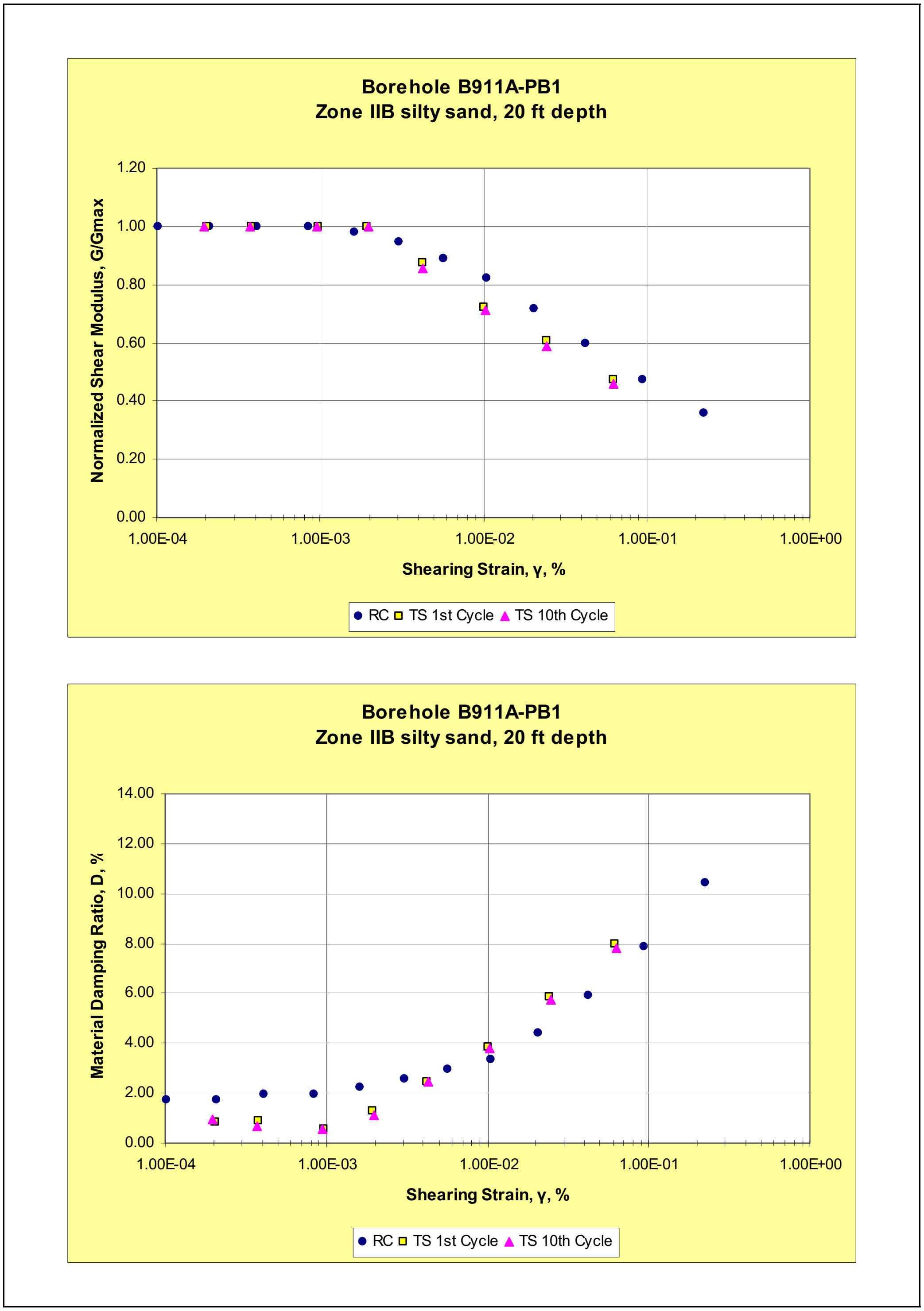


NAPS COL 2.0-29-A **Figure 2.5-223 RCTS Test Results (Sheet 2 of 3) G/G_{max} and D vs. Strain, B-911A UD-1: 5.6 psi Confining Pressure**

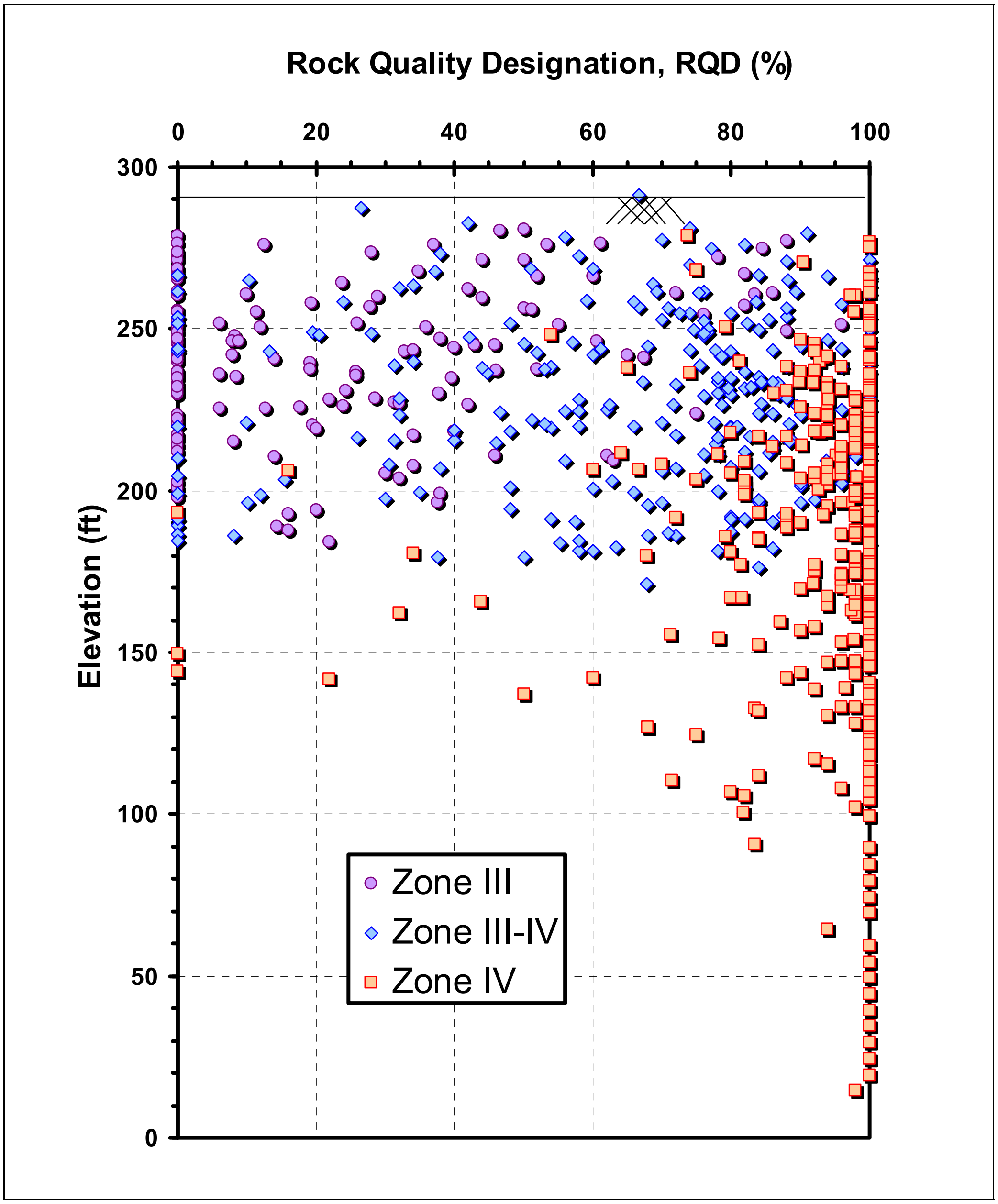




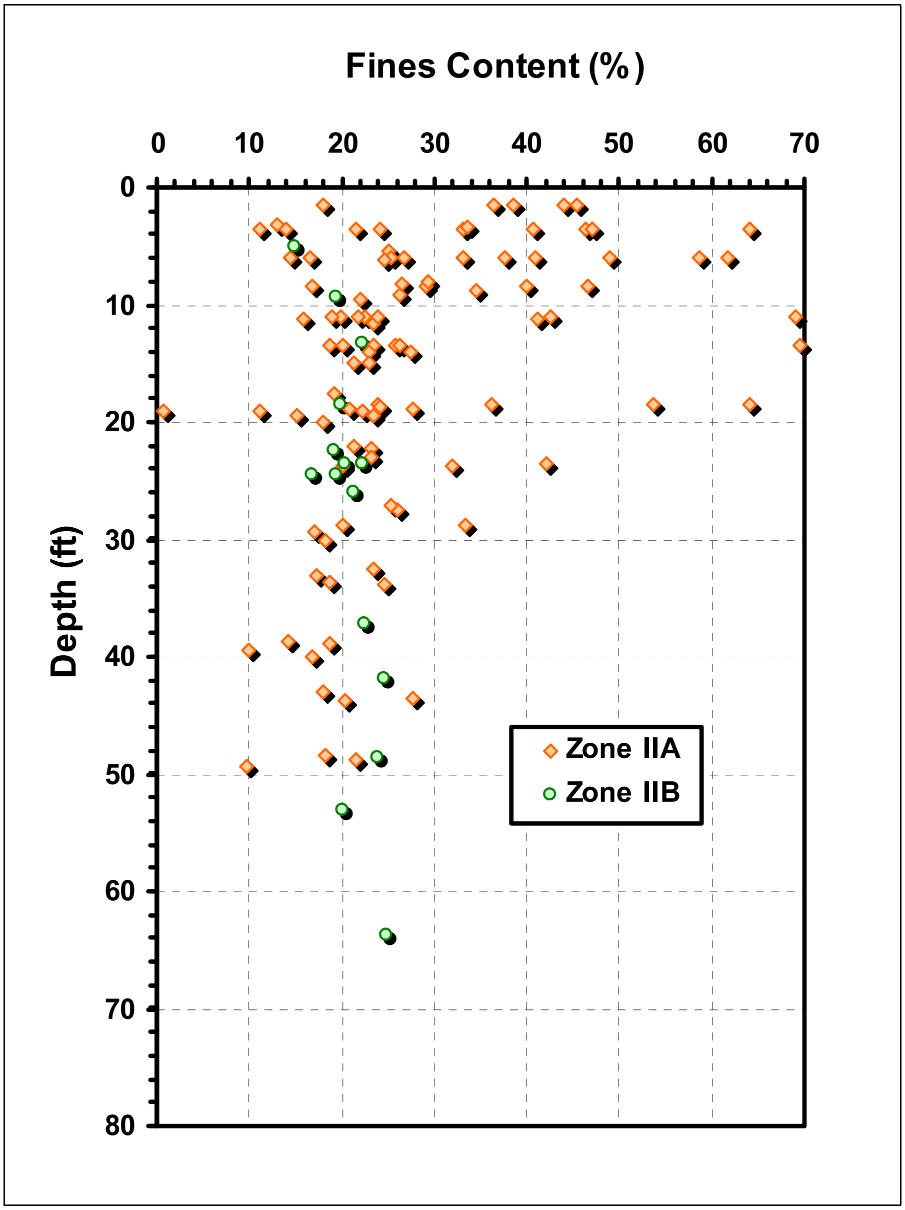
NAPS COL 2.0-29-A **Figure 2.5-223 RCTS Test Results (Sheet 3 of 3) G/G_{max} and D vs. Strain, B-911A PB-1: 11.4 psi Confining Pressure**



NAPS COL 2.0-29-A **Figure 2.5-224 Rock Quality Designation versus Elevation**

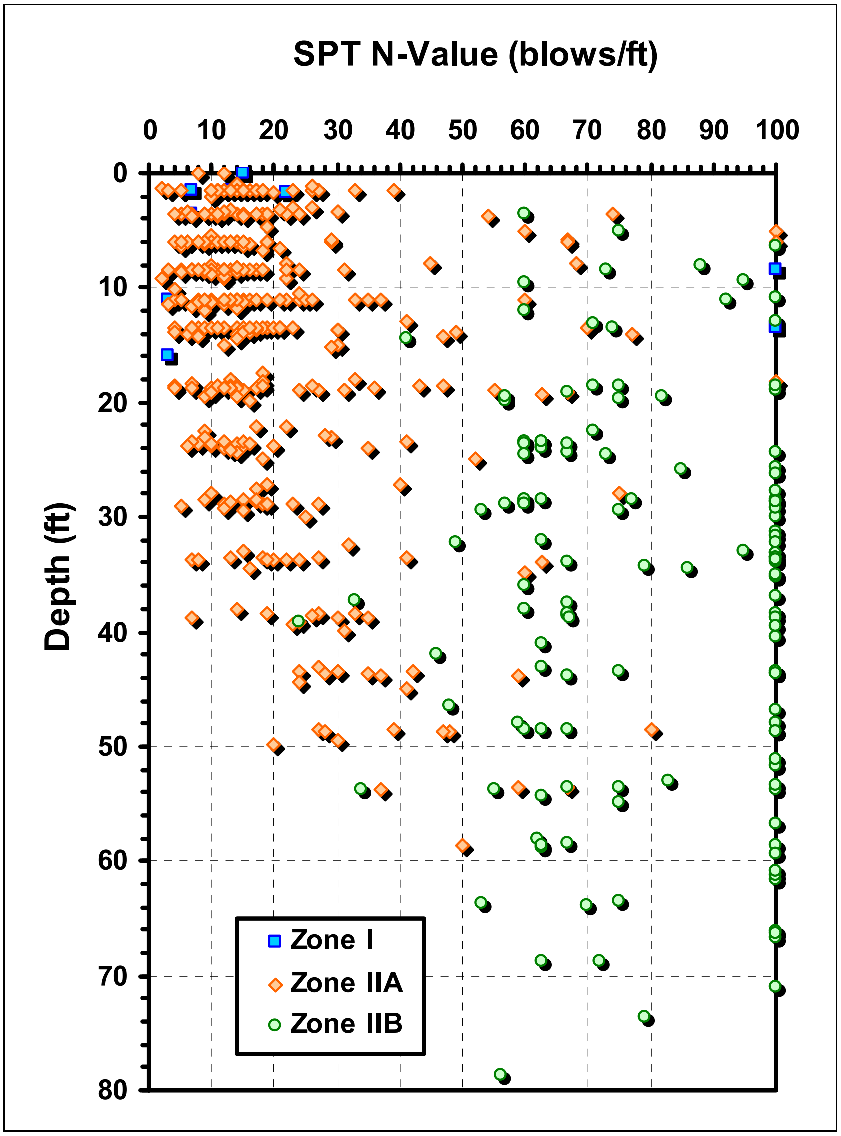


NAPS COL 2.0-29-A **Figure 2.5-225 Fines Content of Saprolite versus Depth**

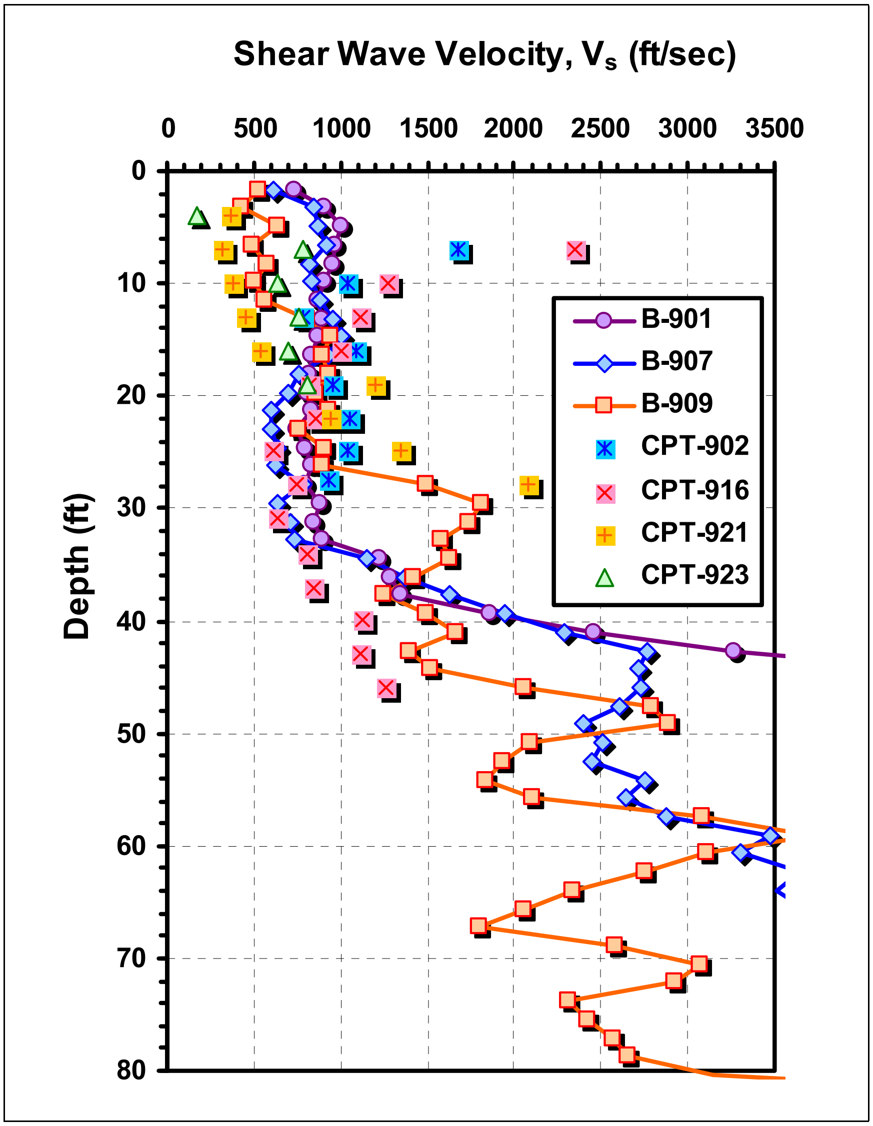


NAPS COL 2.0-29-A

Figure 2.5-226 Measured SPT N-Value versus Depth

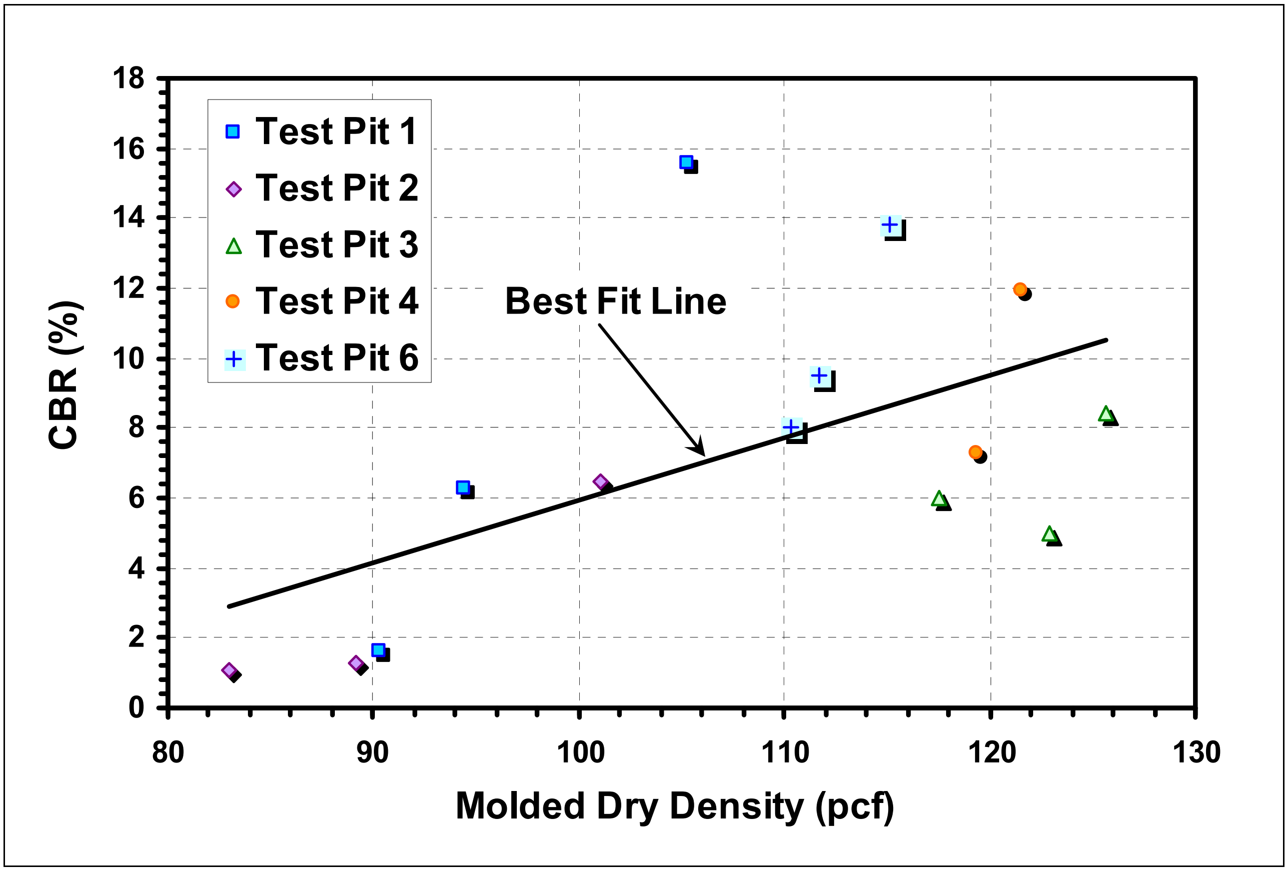


NAPS COL 2.0-29-A **Figure 2.5-227 Measured Soil Shear Wave Velocity versus Depth**



NAPS COL 2.0-29-A

Figure 2.5-228 Relationship between CBR and Molded Dry Density



NAPS ESP COL 2.5-3 **Figure 2.5-229 Cross-Section A-A'**

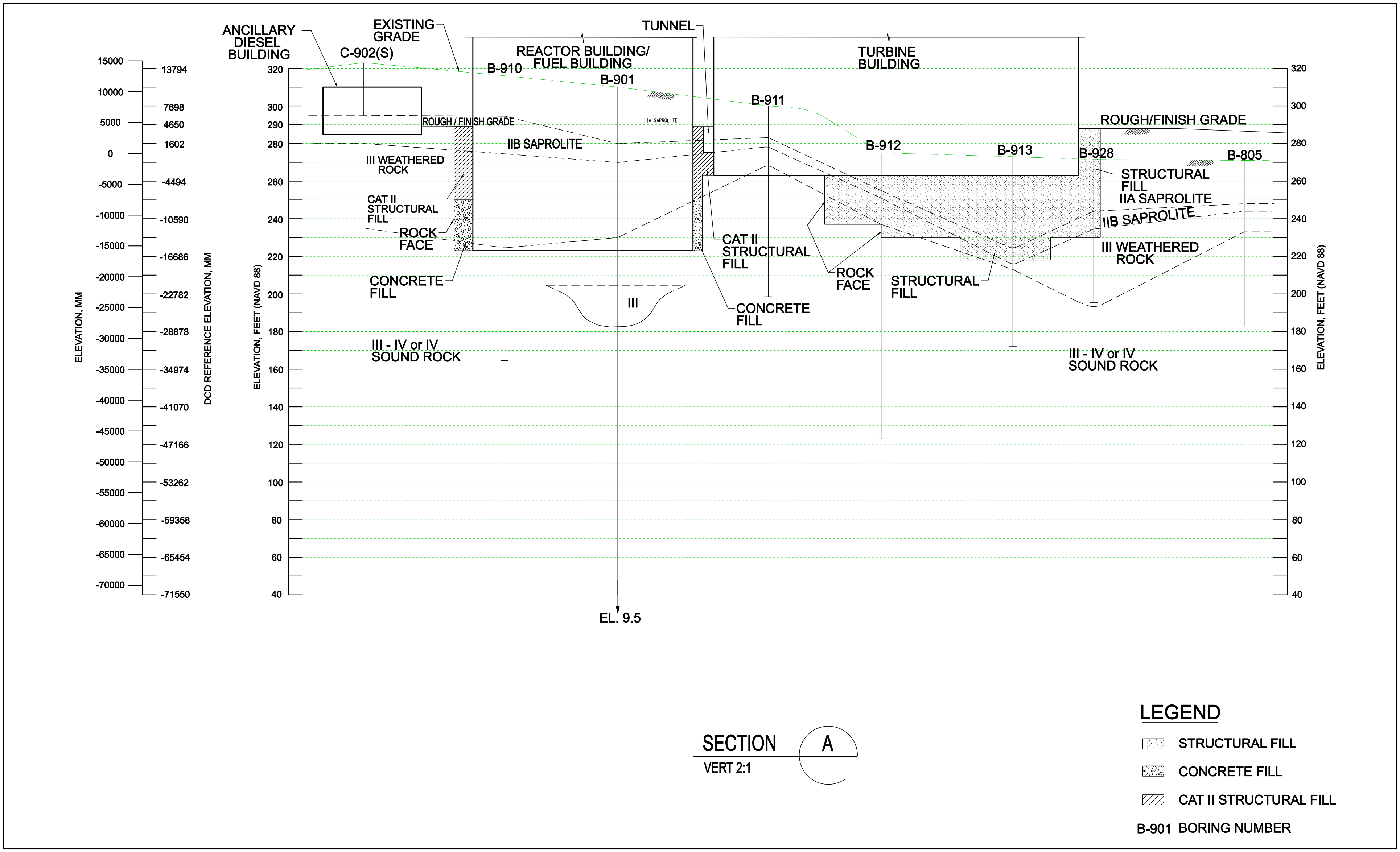
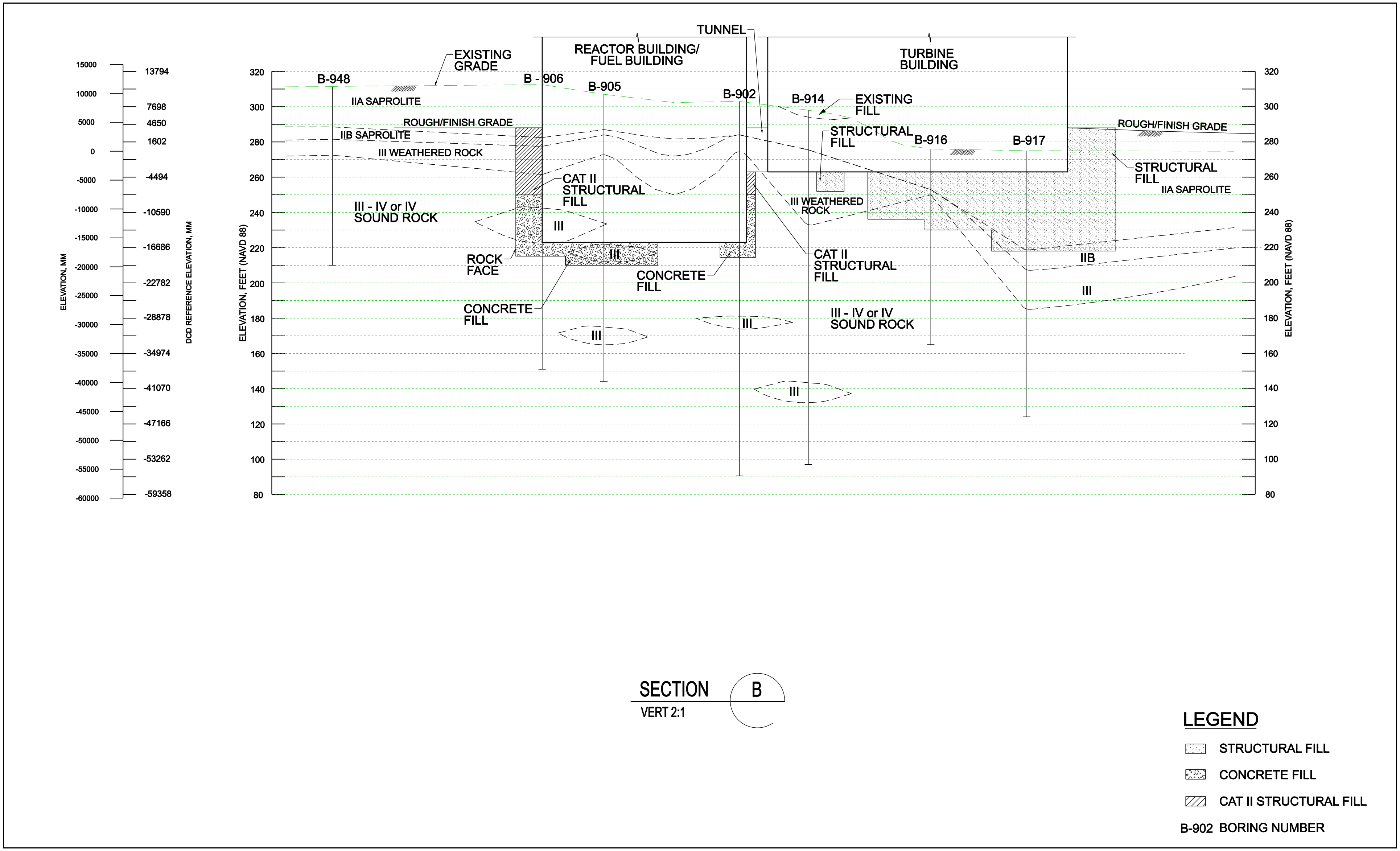
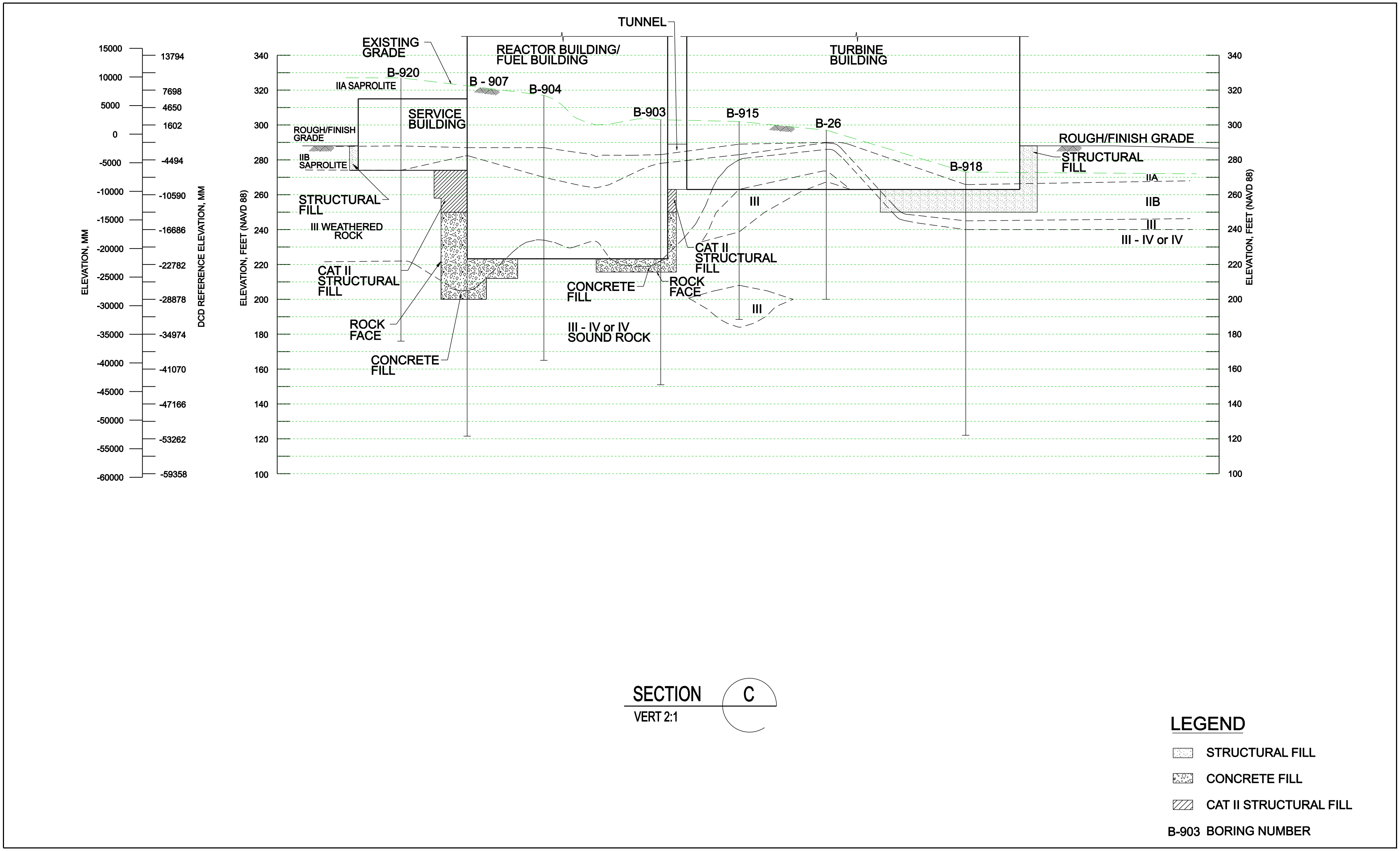


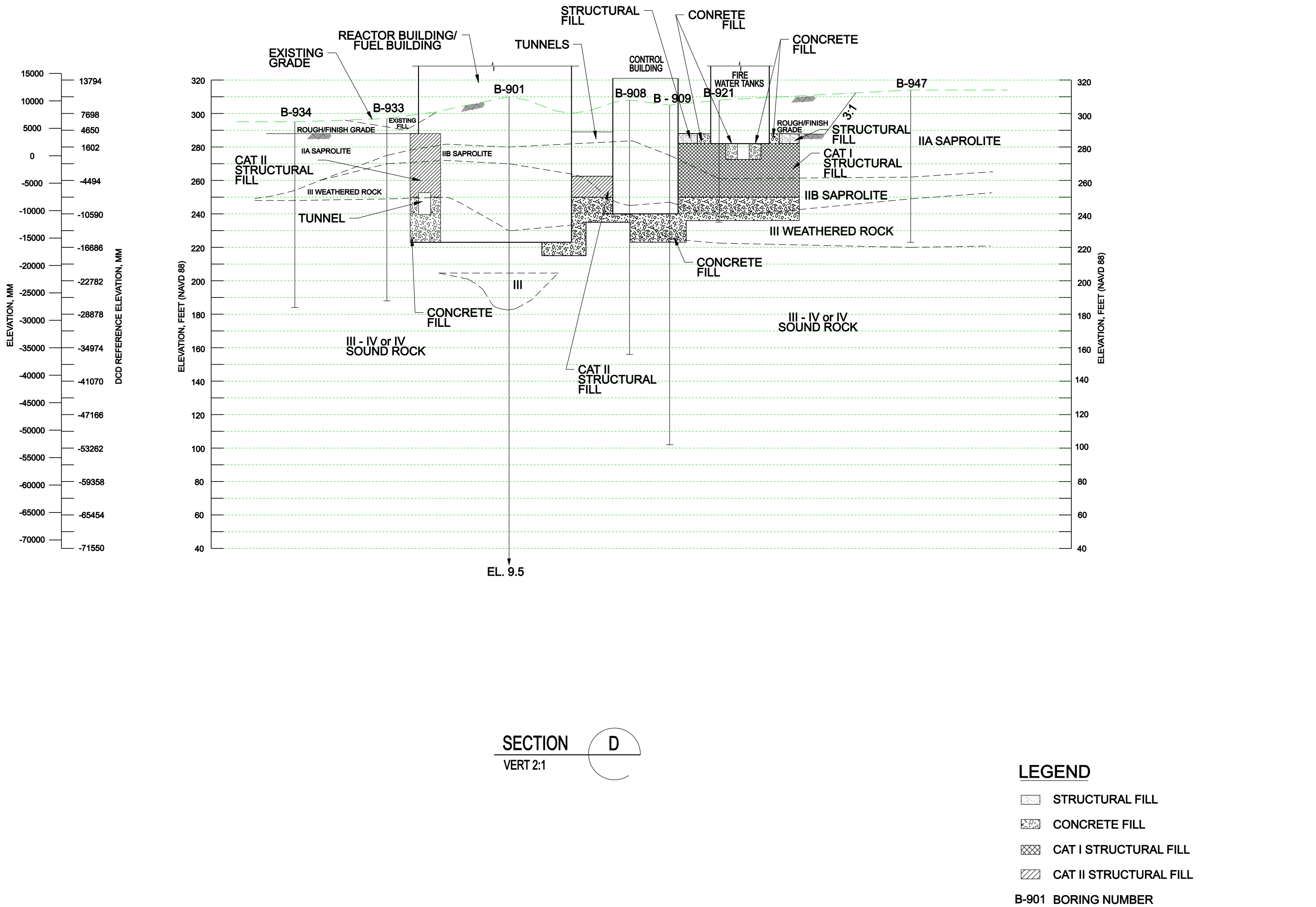
Figure 2.5-230 Cross-Section B-B'



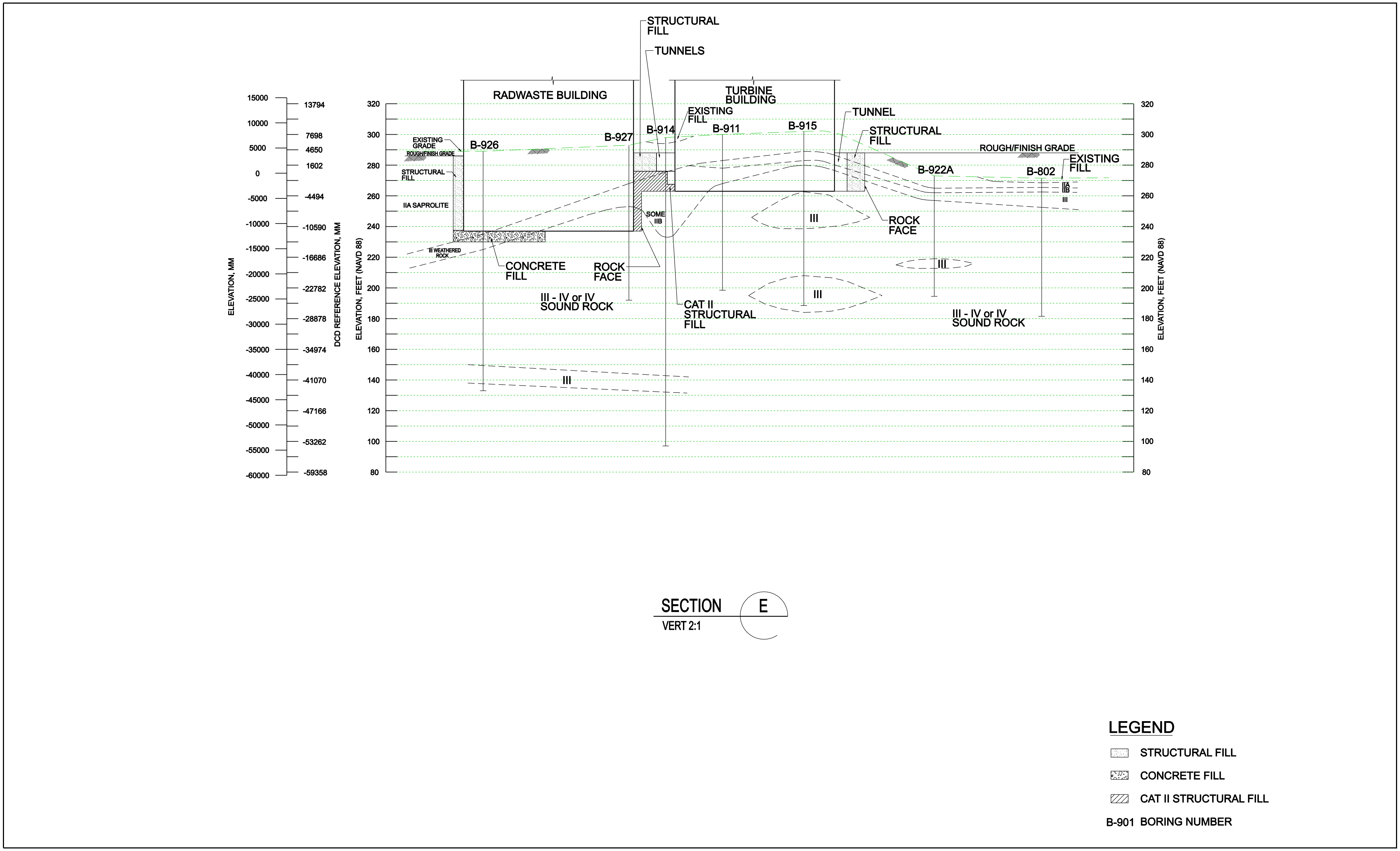
NAPS ESP COL 2.5-3 **Figure 2.5-231 Cross-Section C-C'**



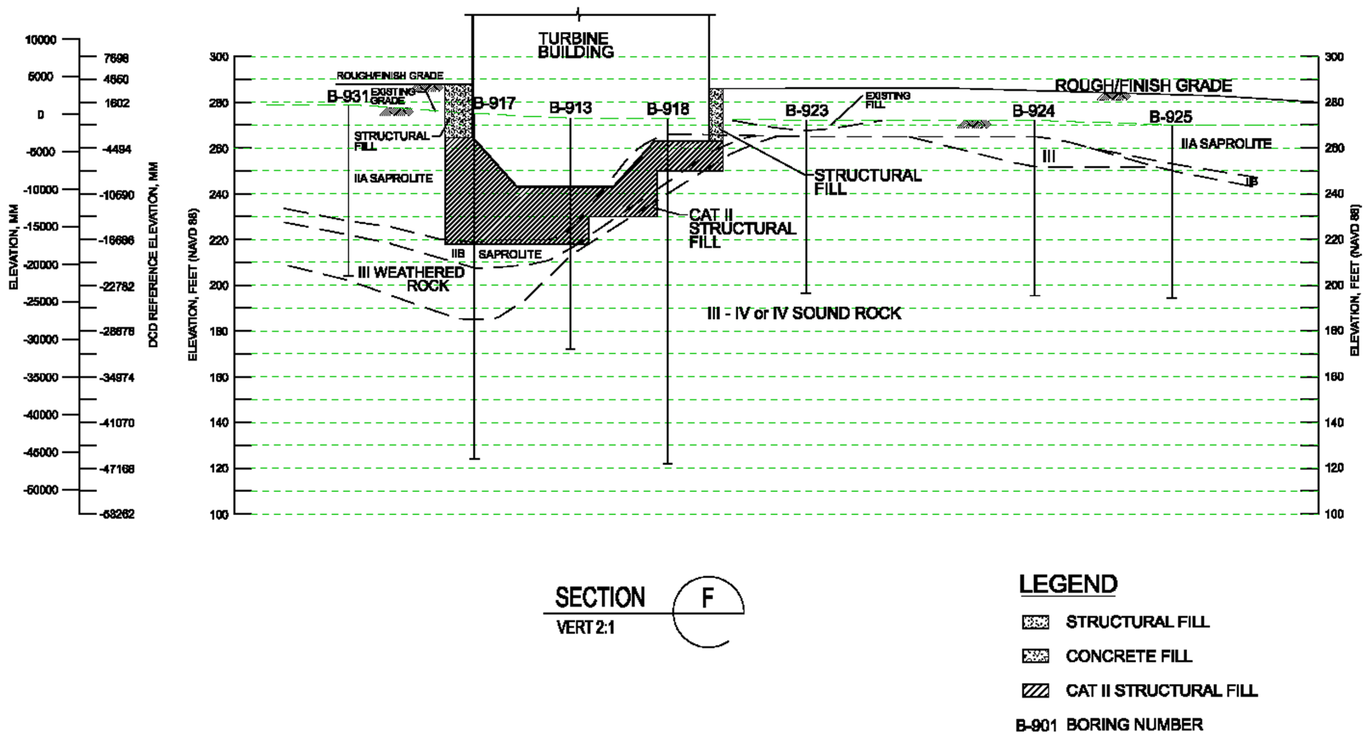
NAPS ESP COL 2.5-3 **Figure 2.5-232 Cross-Section D-D'**



NAPS ESP COL 2.5-3 **Figure 2.5-233 Cross-Section E-E'**

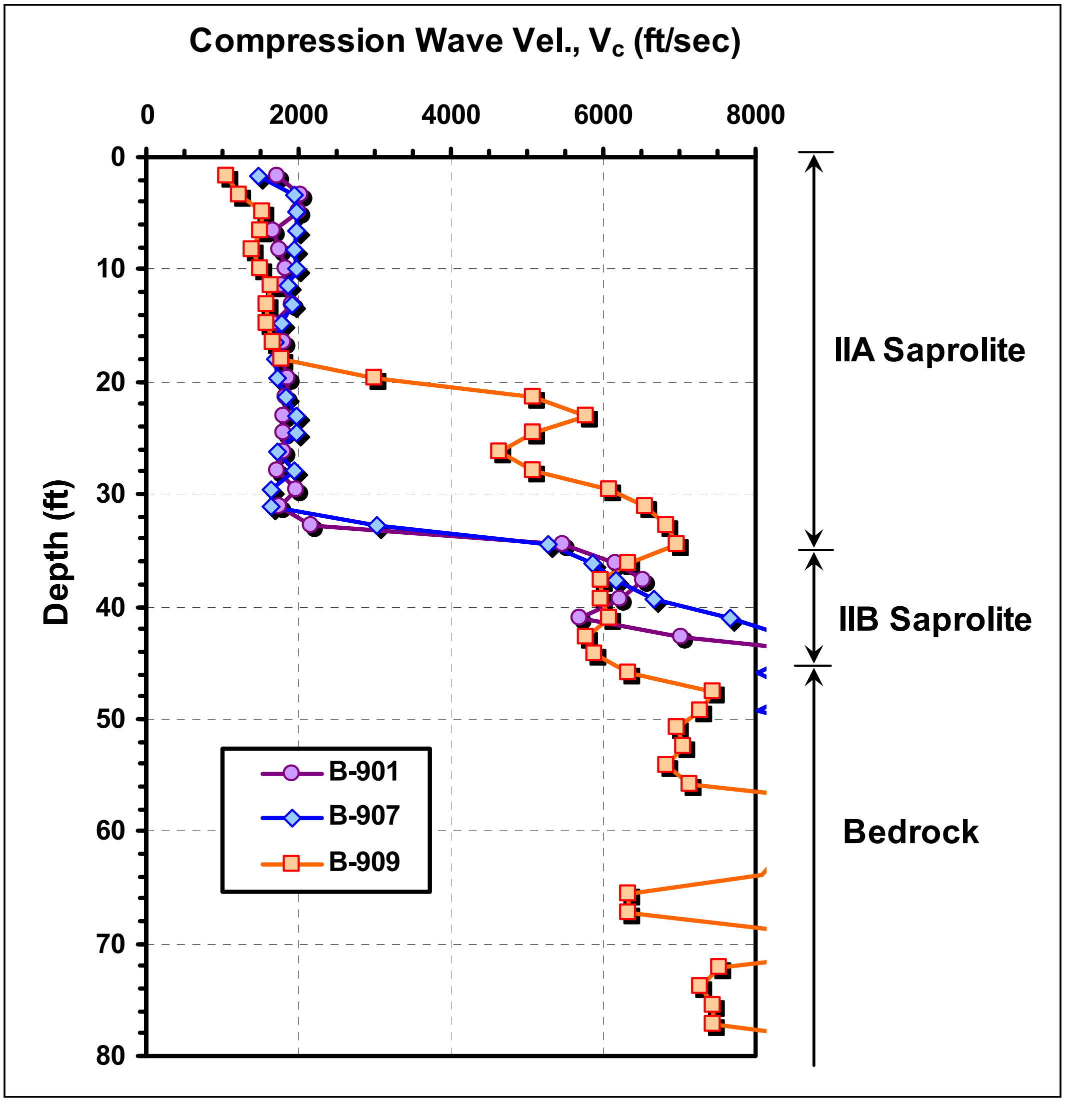


NAPS ESP COL 2.5-3 **Figure 2.5-234 Cross-Section F-F'**



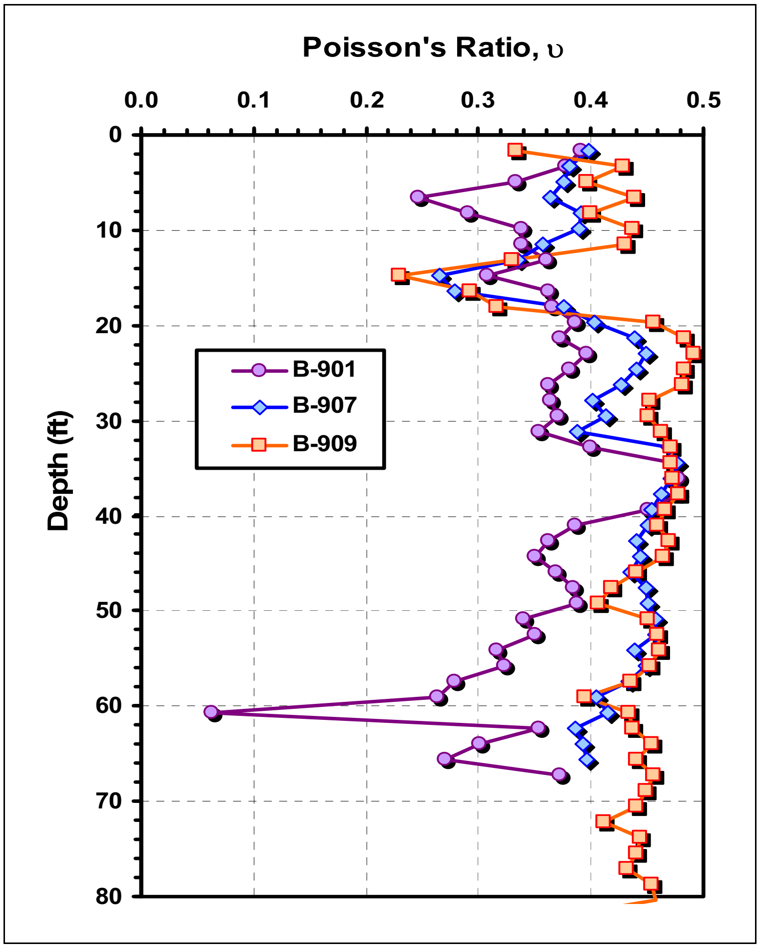
NAPS COL 2.0-29-A

Figure 2.5-235 Measured Compression Wave Velocity versus Depth

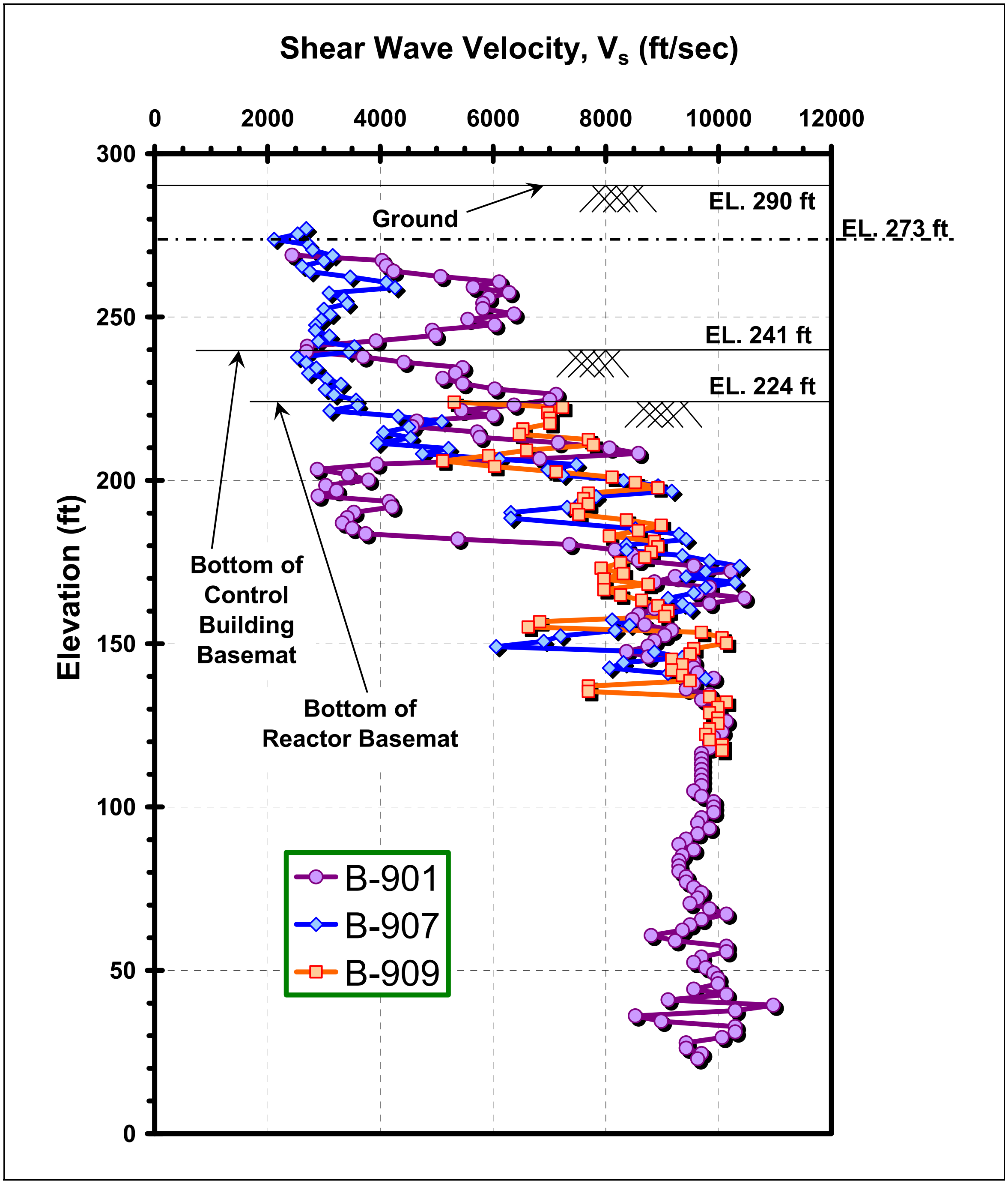


NAPS COL 2.0-29-A

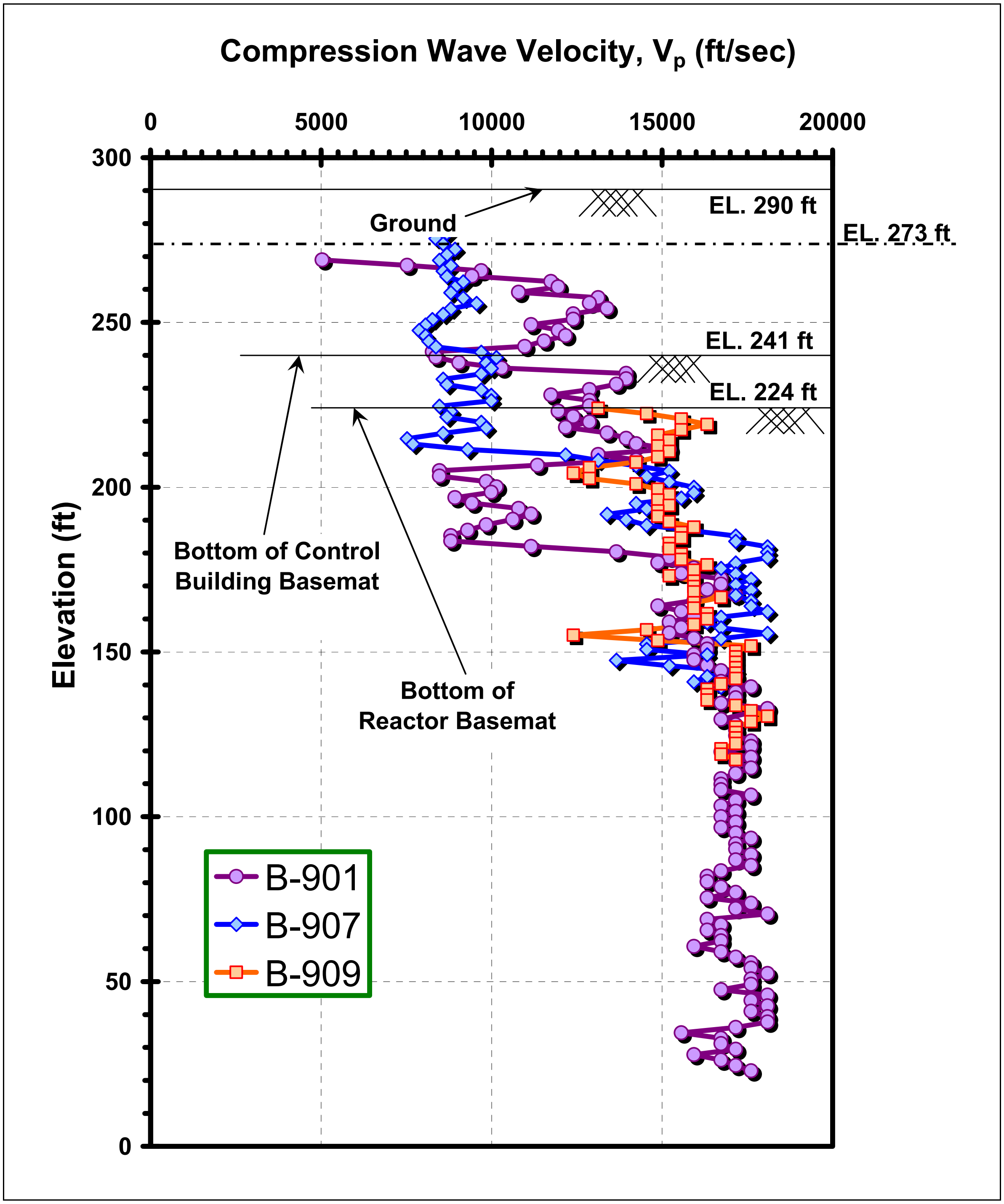
Figure 2.5-236 Soil Poisson's Ratio versus Depth



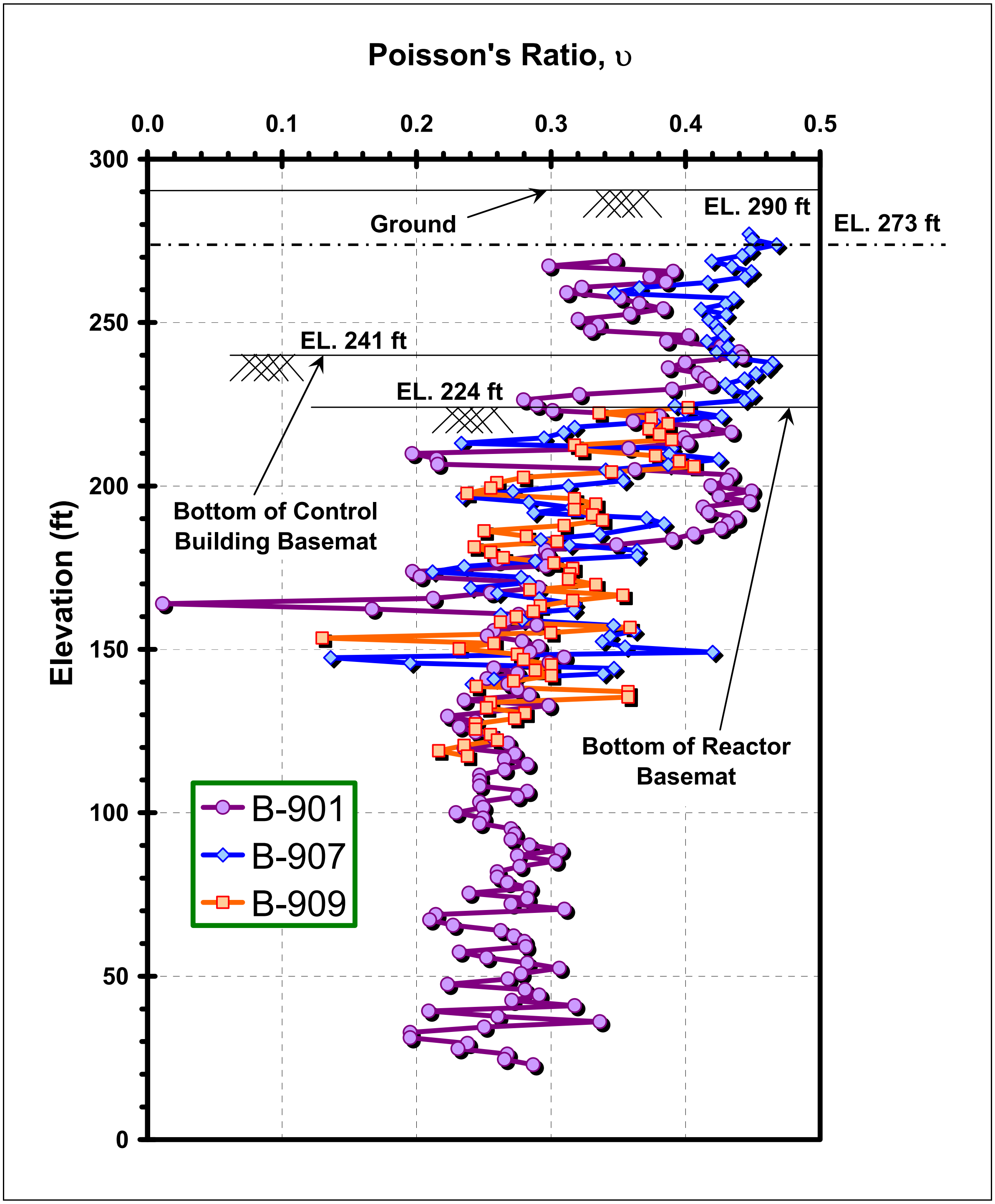
NAPS COL 2.0-29-A **Figure 2.5-237 Bedrock Shear Wave Velocity versus Elevation**



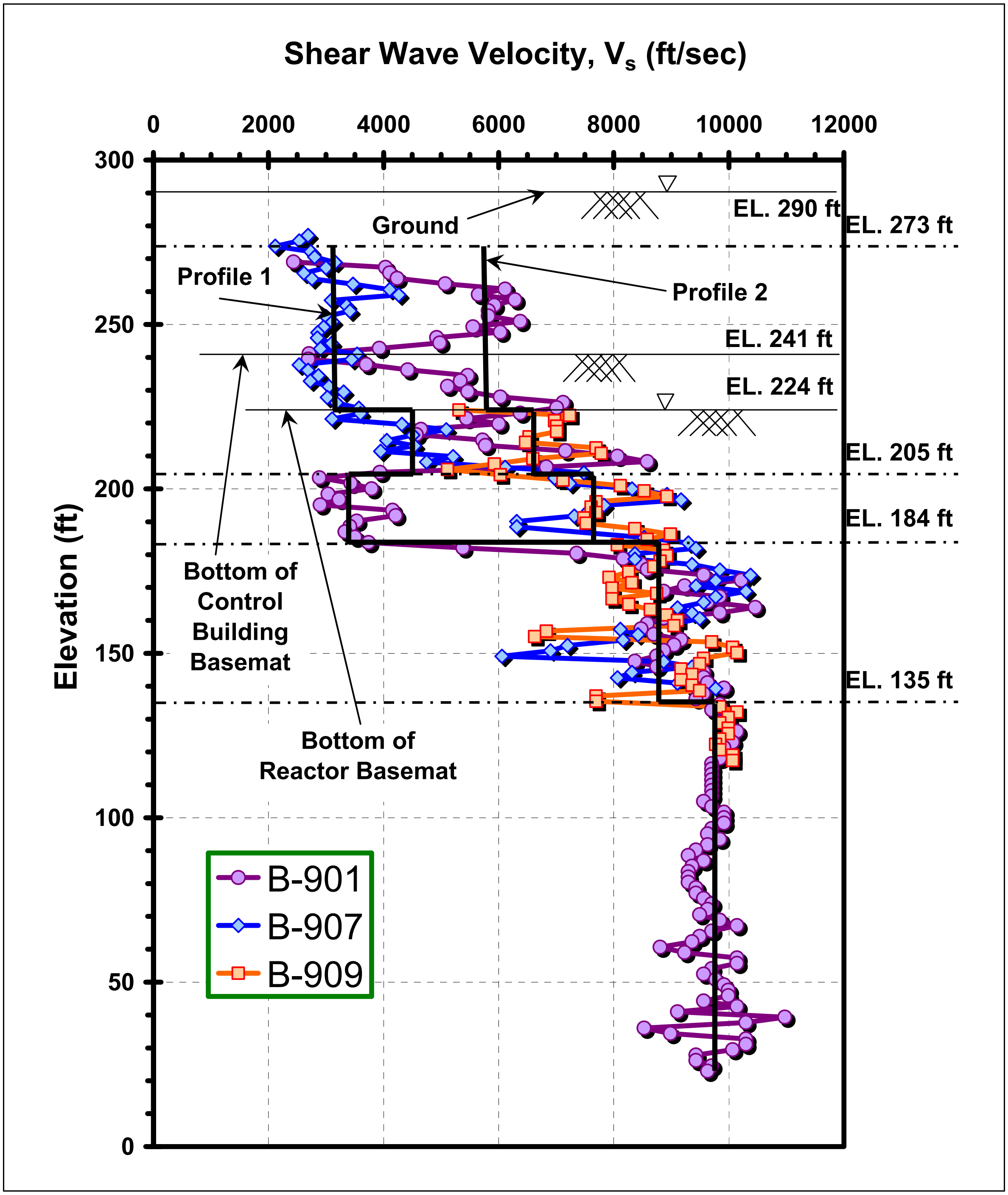
NAPS COL 2.0-29-A **Figure 2.5-238 Bedrock Compression Wave Velocity versus Elevation**



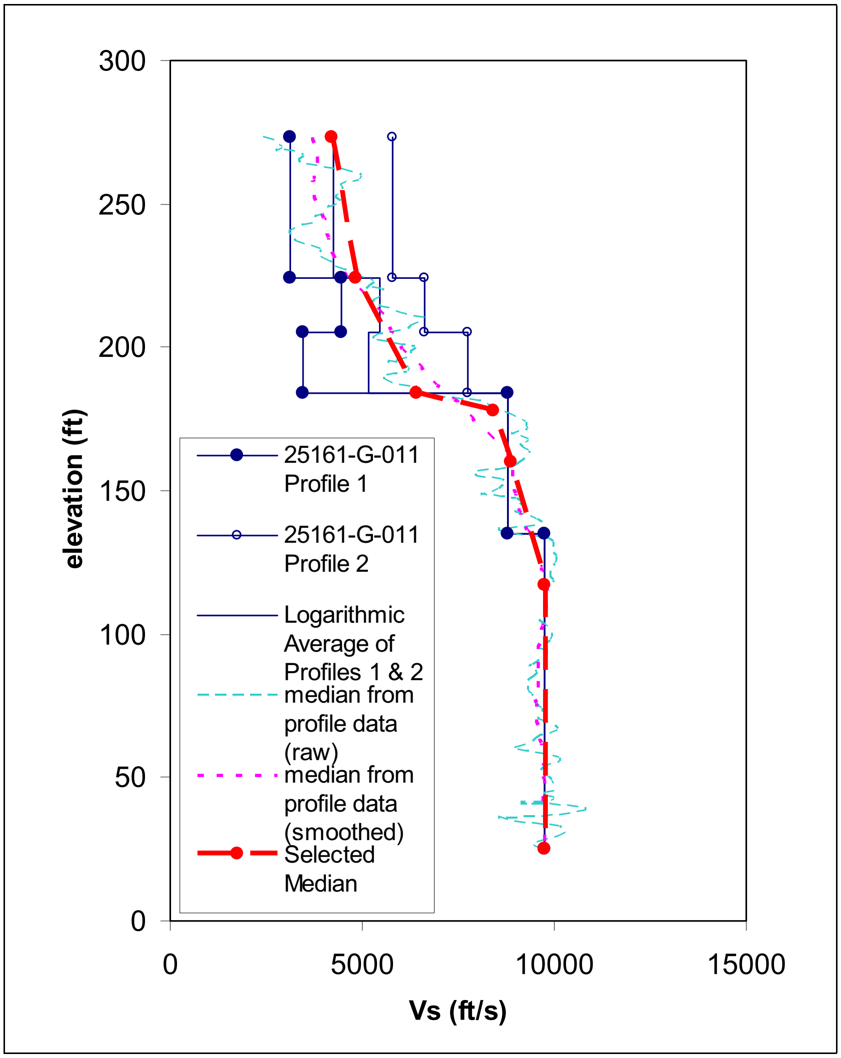
NAPS COL 2.0-29-A **Figure 2.5-239 Bedrock Poisson's Ratio versus Elevation**



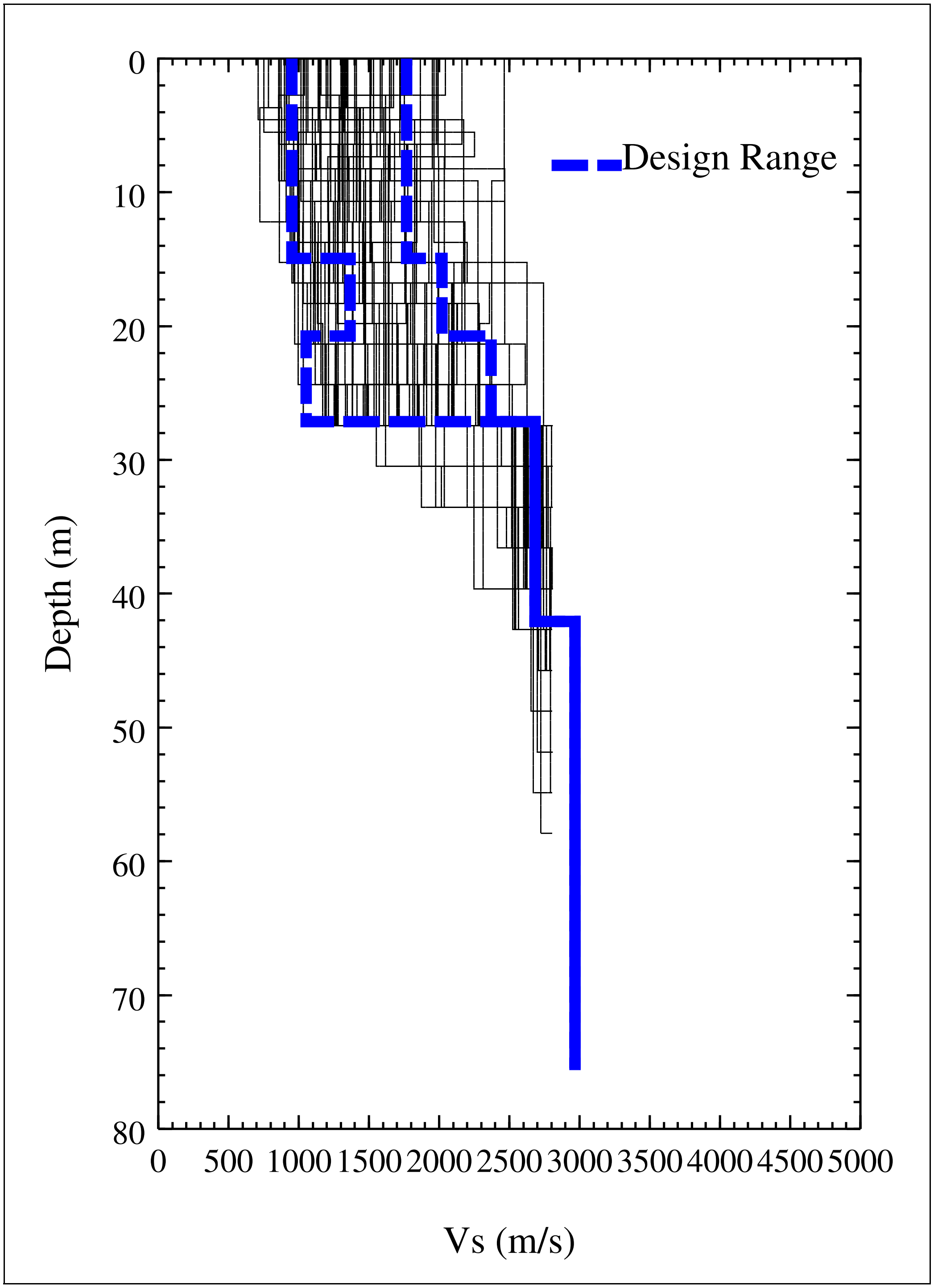
NAPS ESP COL 2.5-9 **Figure 2.5-240 Design Bedrock Shear Wave Velocity versus Elevation**



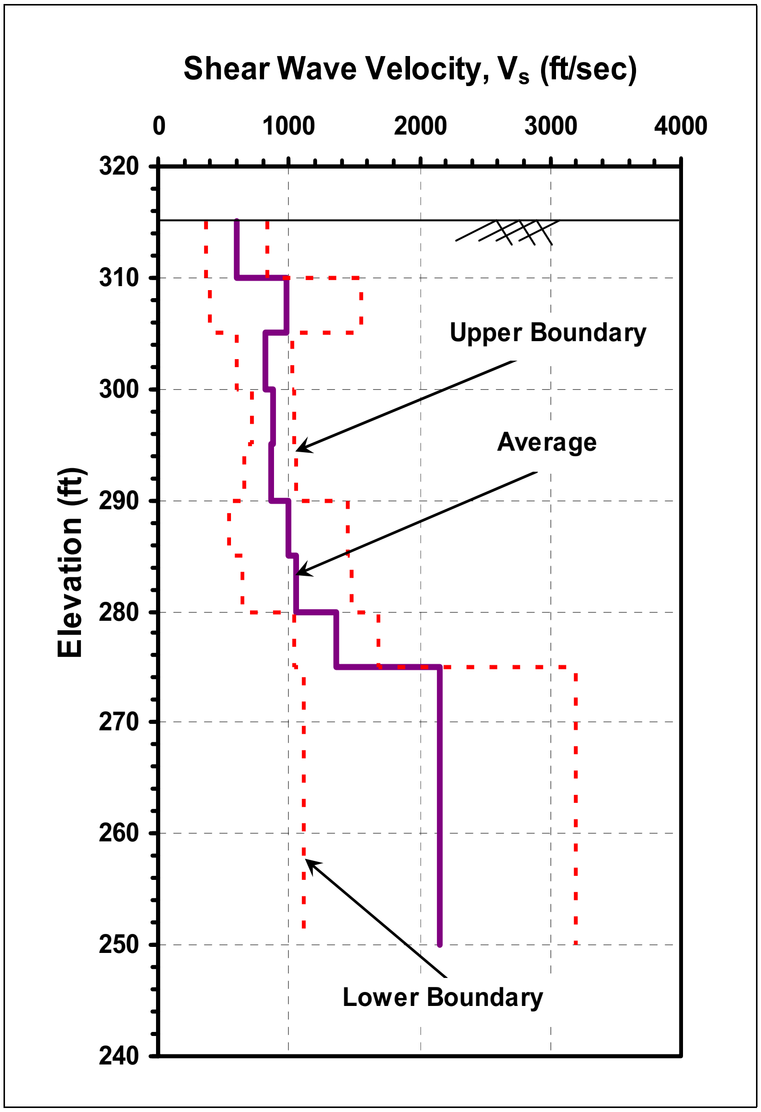
NAPS ESP COL 2.5-9 **Figure 2.5-241 Median Shear Wave Velocity versus Depth**



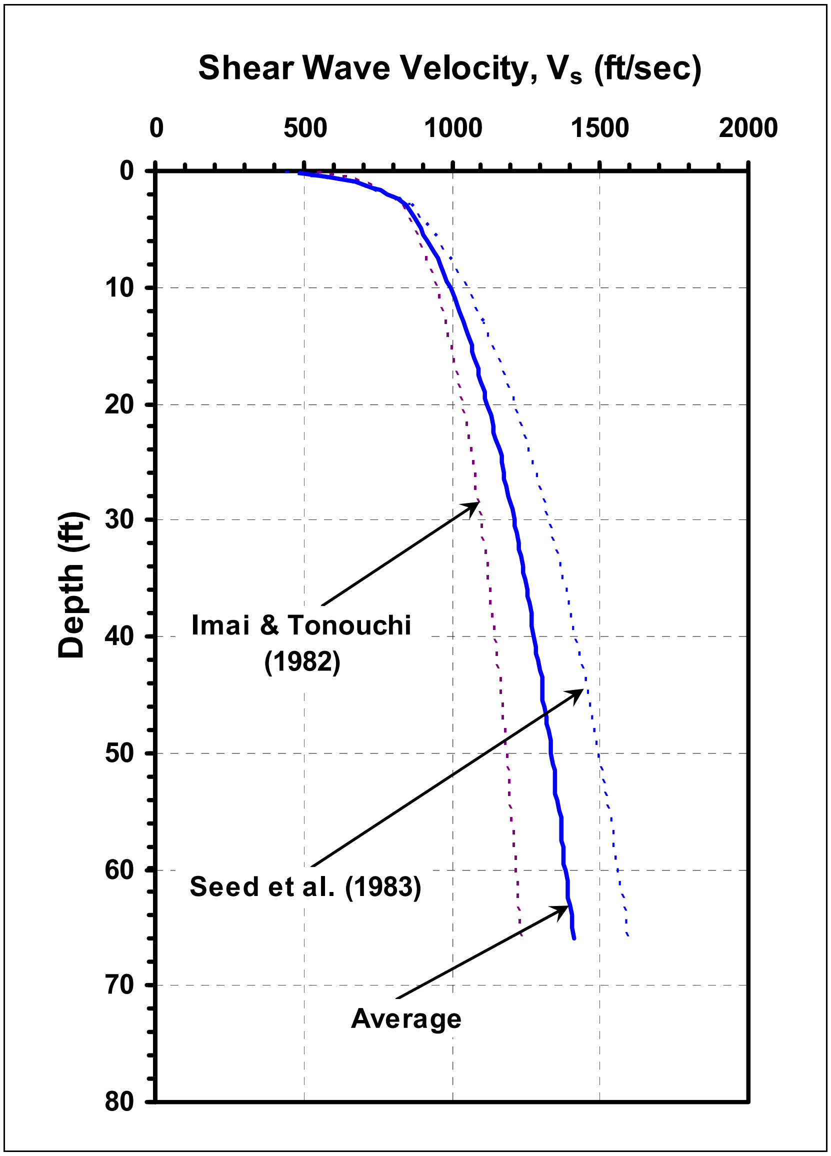
NAPS ESP COL 2.5-9 **Figure 2.5-242 Randomized Rock Shear Wave Velocity Profiles**



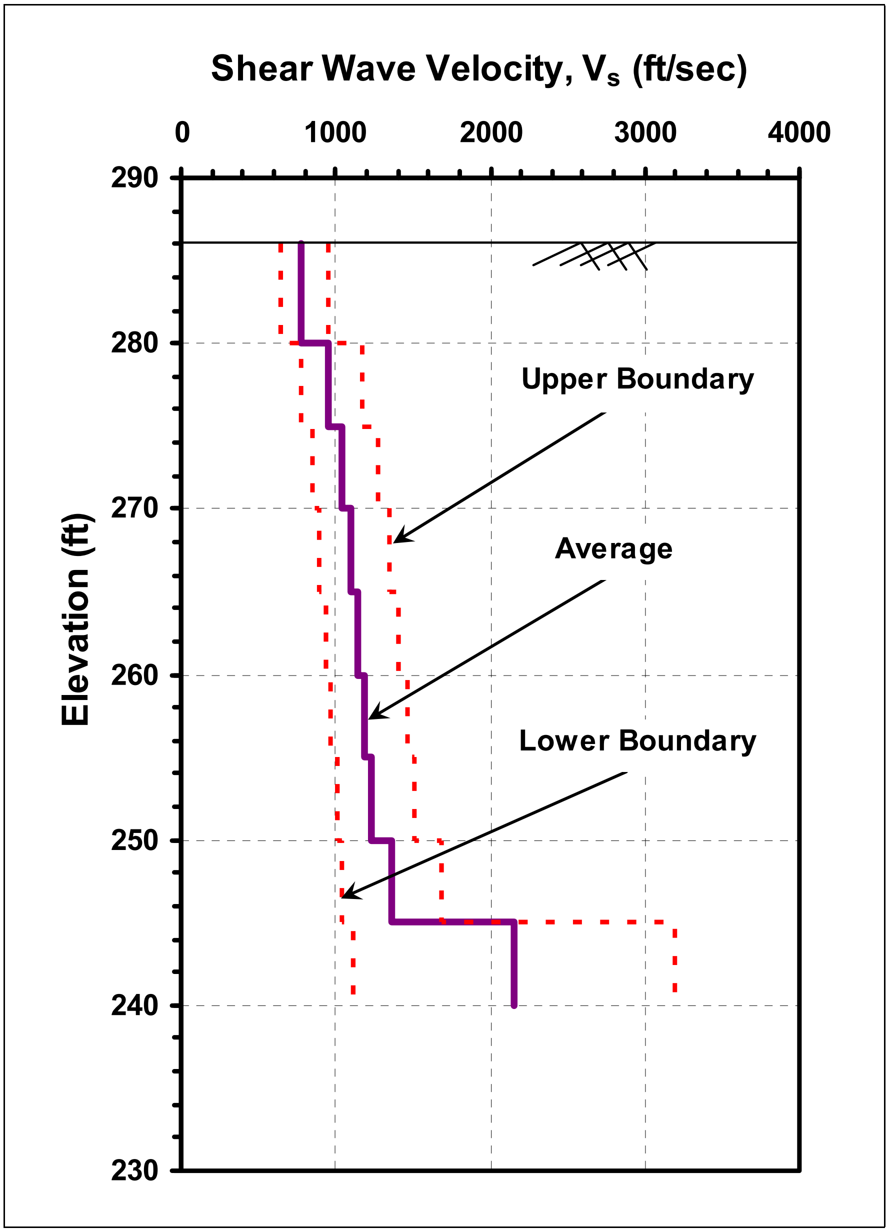
NAPS ESP COL 2.5-9 **Figure 2.5-243 Shear Wave Velocity versus Elevation for In-Situ Soils Averaged Over 5-Foot Intervals**



NAPS ESP COL 2.5-9 **Figure 2.5-244 Estimated Shear Wave Velocity versus Depth for Structural Fill**

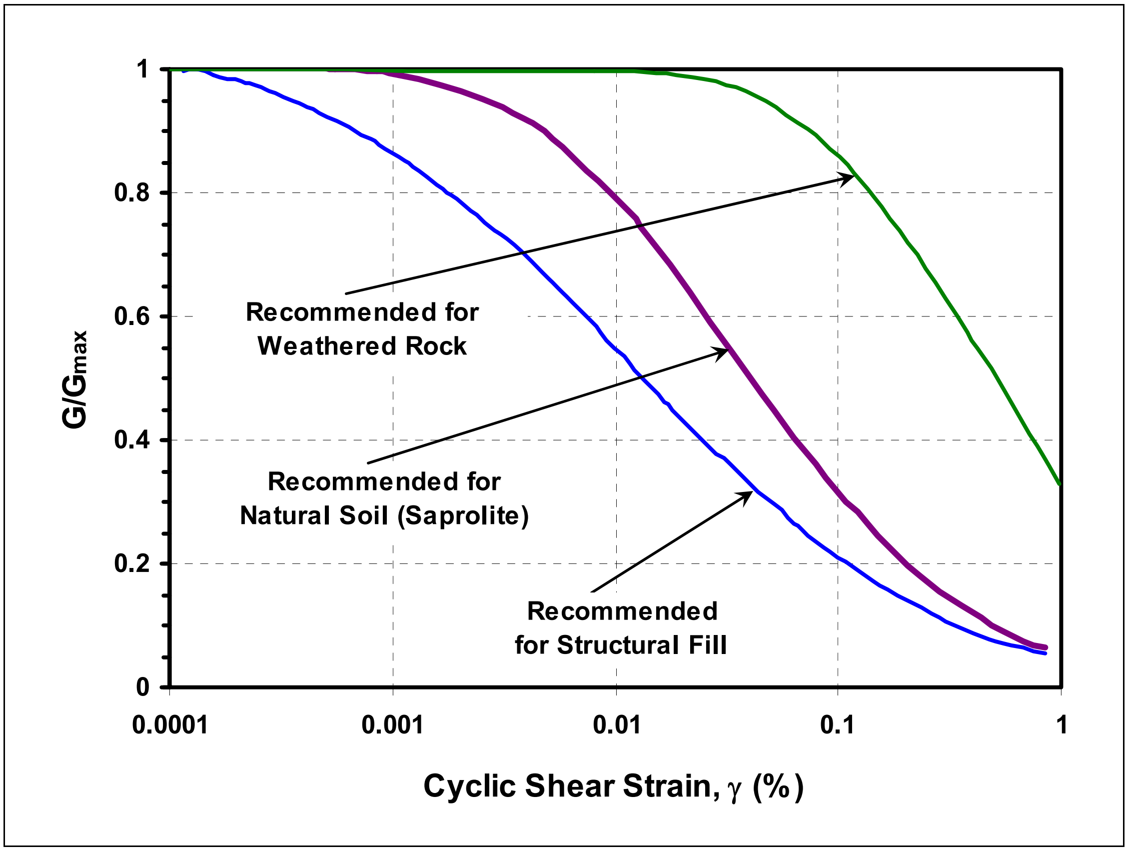


NAPS ESP COL 2.5-9 **Figure 2.5-245 Shear Wave Velocity versus Elevation for Structural Fill Averaged Over 5-Foot Intervals**



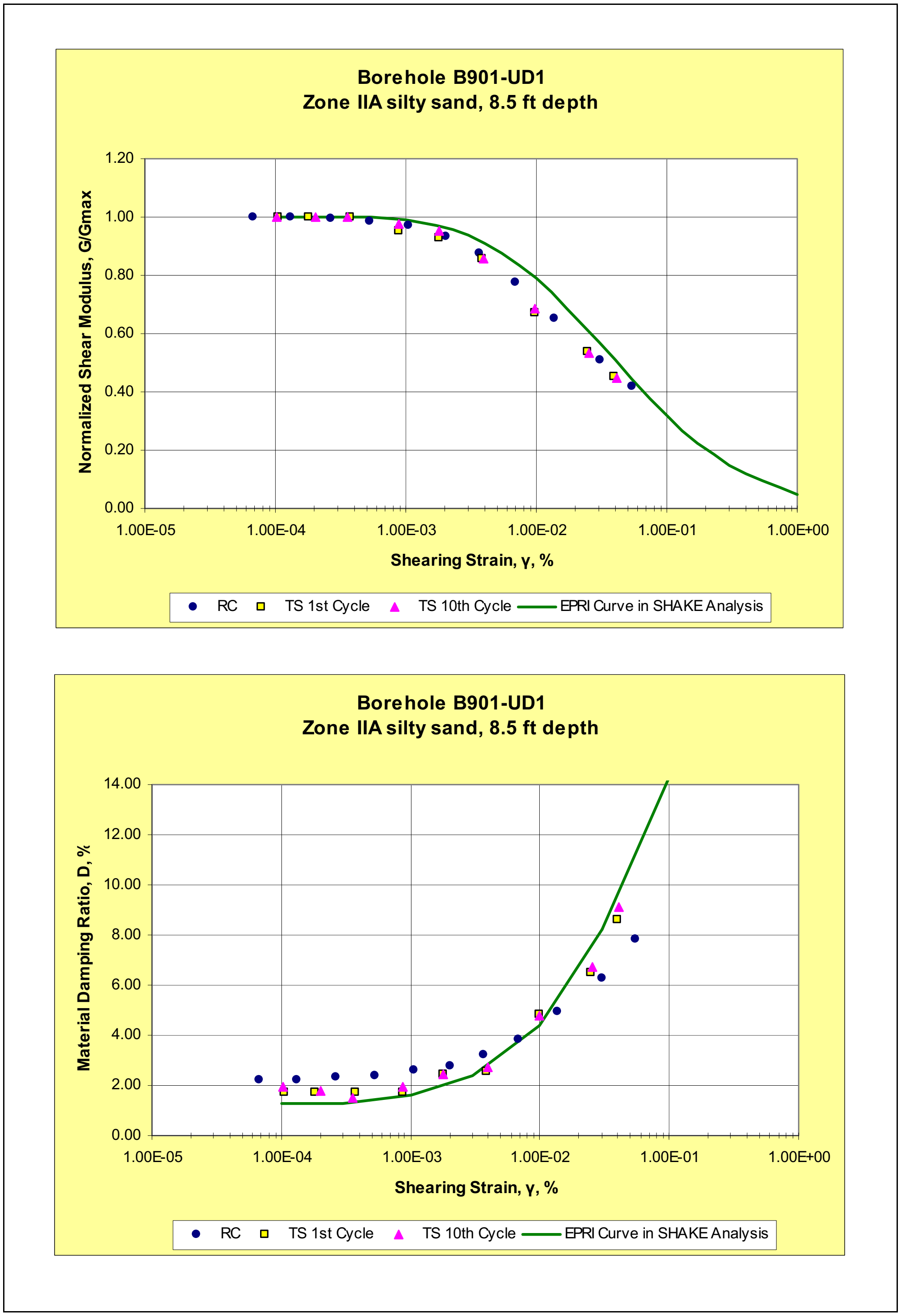
NAPS COL 2.0-29-A

Figure 2.5-246 Shear Modulus Reduction Design Curves



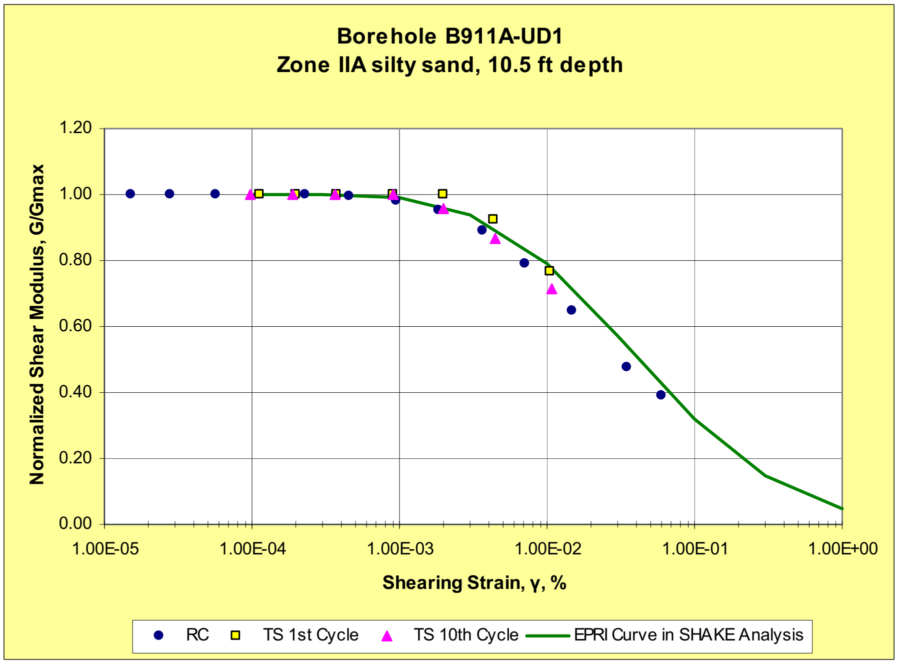
NAPS COL 2.0-29-A

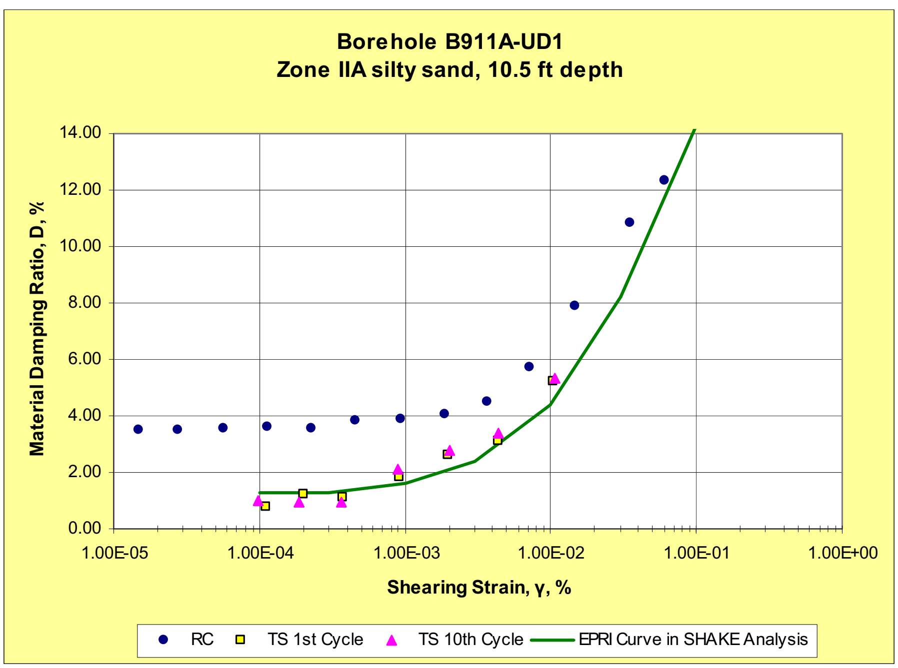
Figure 2.5-247 RCTS Results with G/G_{max} and D Curve G/G_{max} vs. Strain, B-901 UD-1: 4.3 psi Confining Pressure (Sheet 1 of 3)



NAPS COL 2.0-29-A

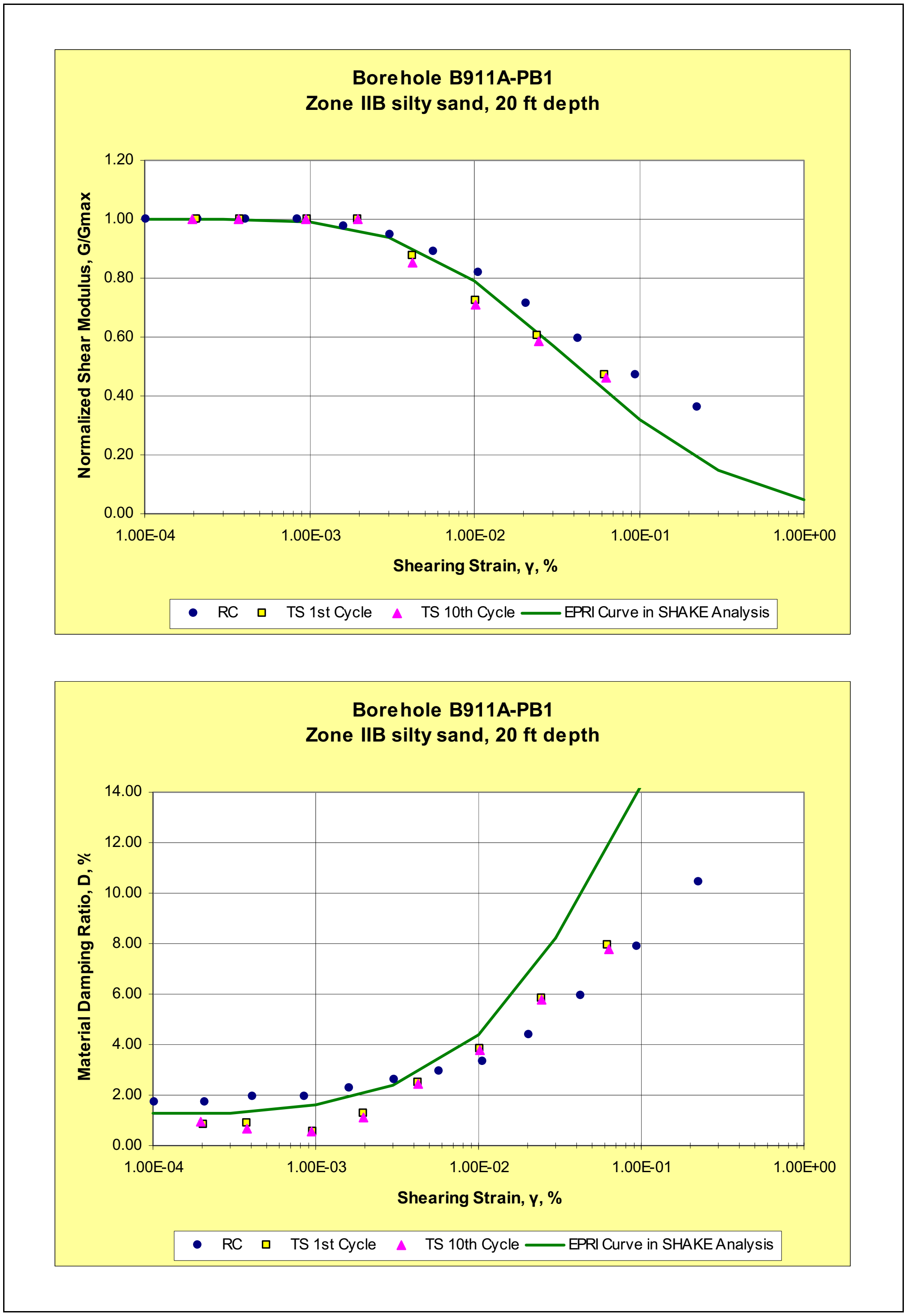
Figure 2.5-247 RCTS Results with G/G_{max} and D Curve G/G_{max} vs. Strain, B-911A UD-1: 5.6 psi Confining Pressure (Sheet 2 of 3)





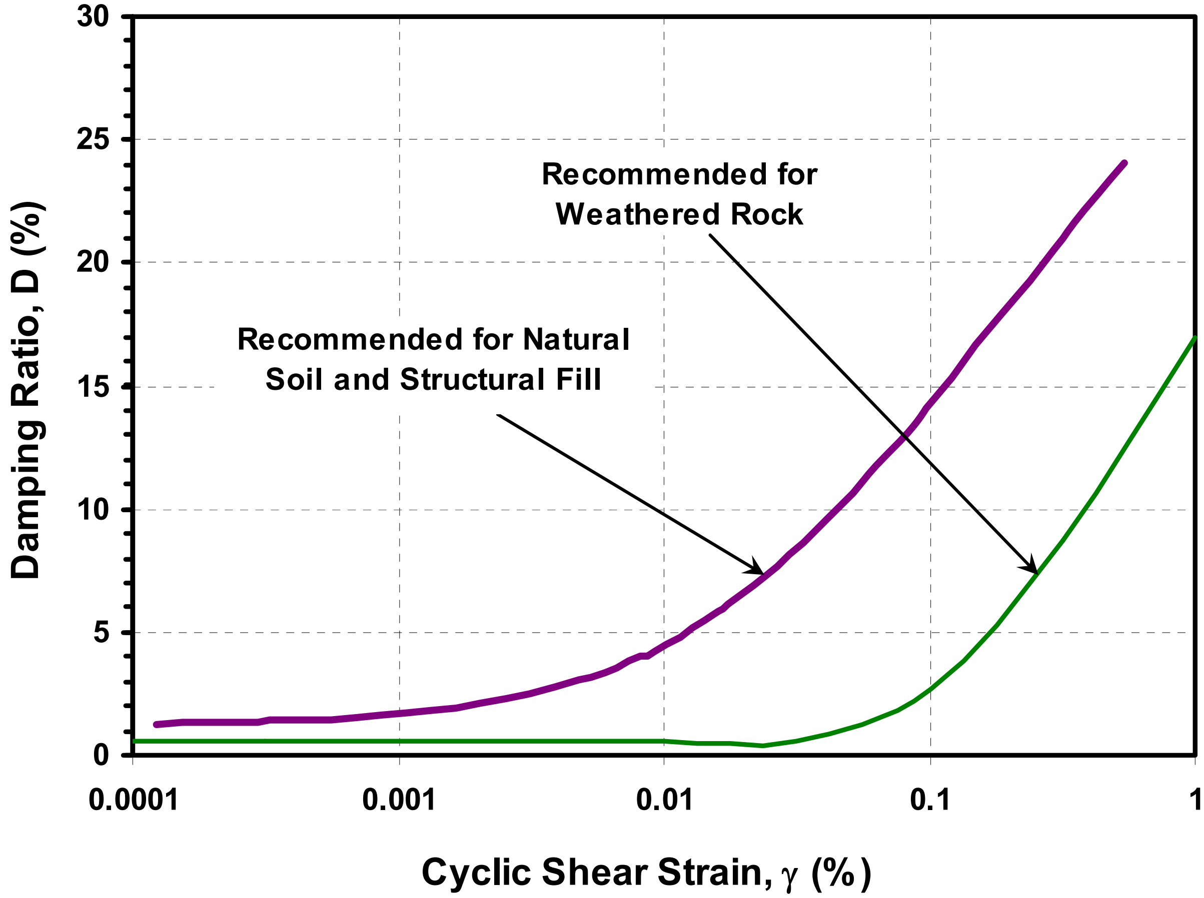
NAPS COL 2.0-29-A

Figure 2.5-247 RCTS Results with G/G_{max} and D Curve G/G_{max} vs. Strain, B-911A PB-1: 11.4 psi Confining Pressure (Sheet 3 of 3)

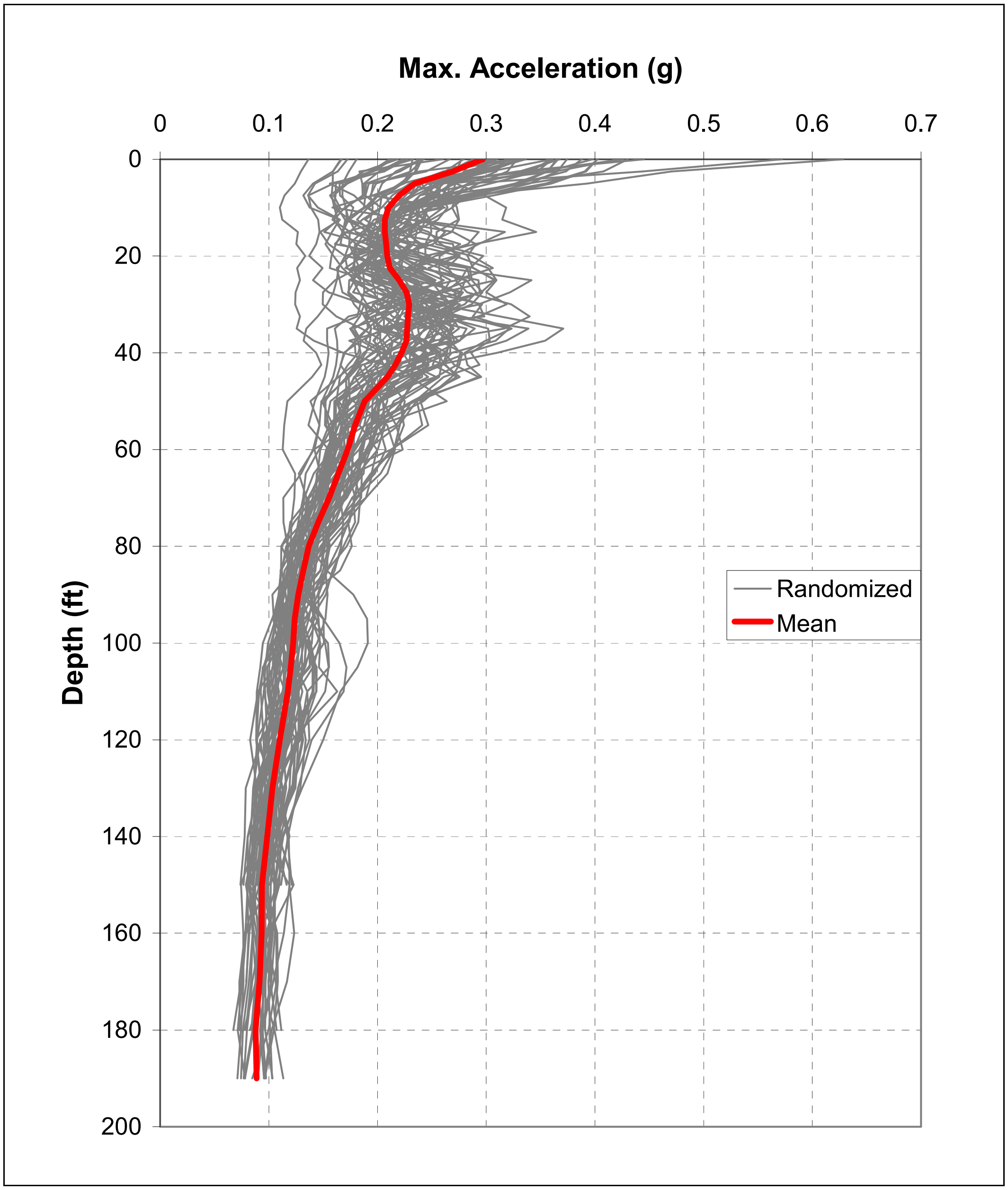


NAPS COL 2.0-29-A

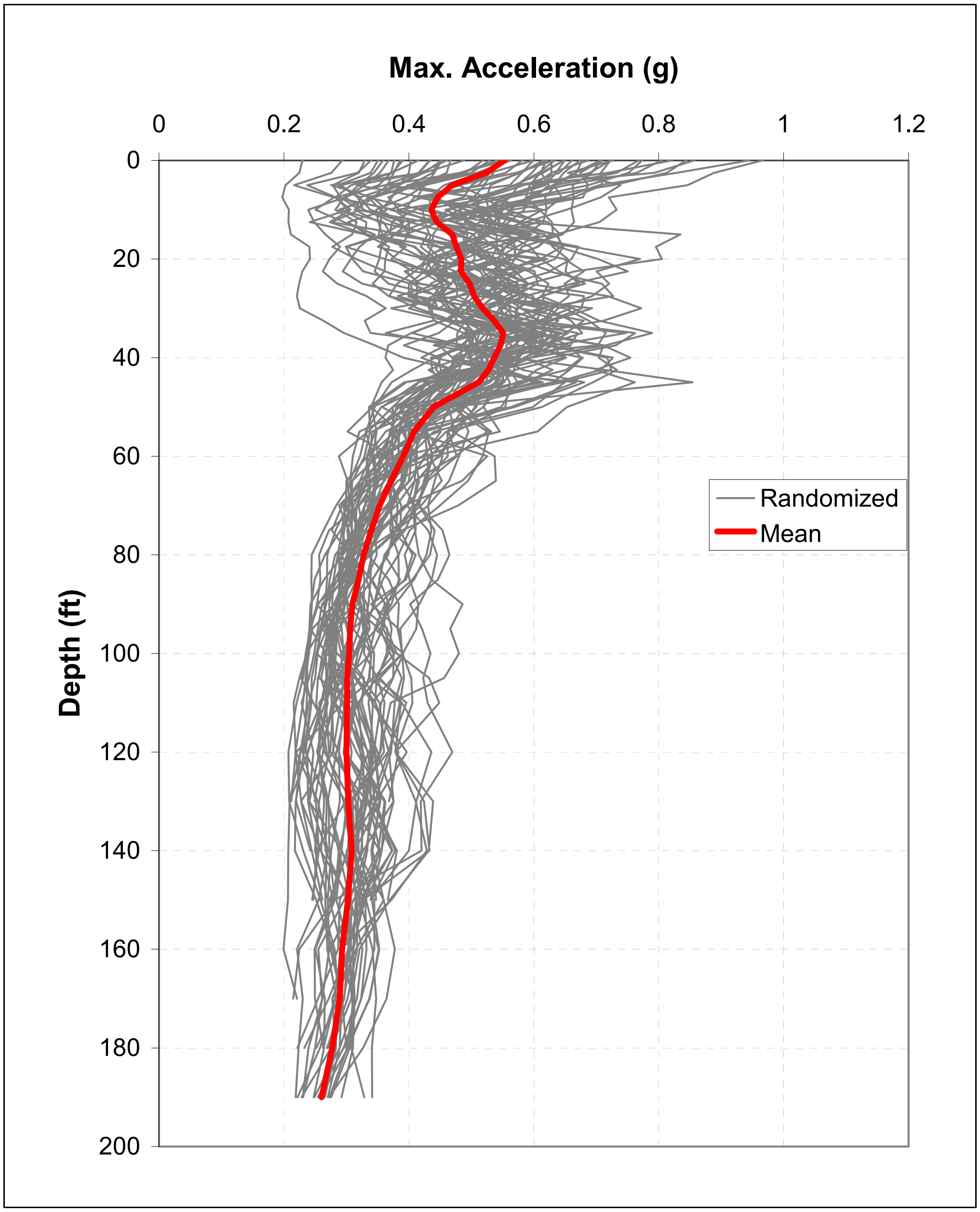
Figure 2.5-248 Damping Ratio versus Cyclic Shear Strain



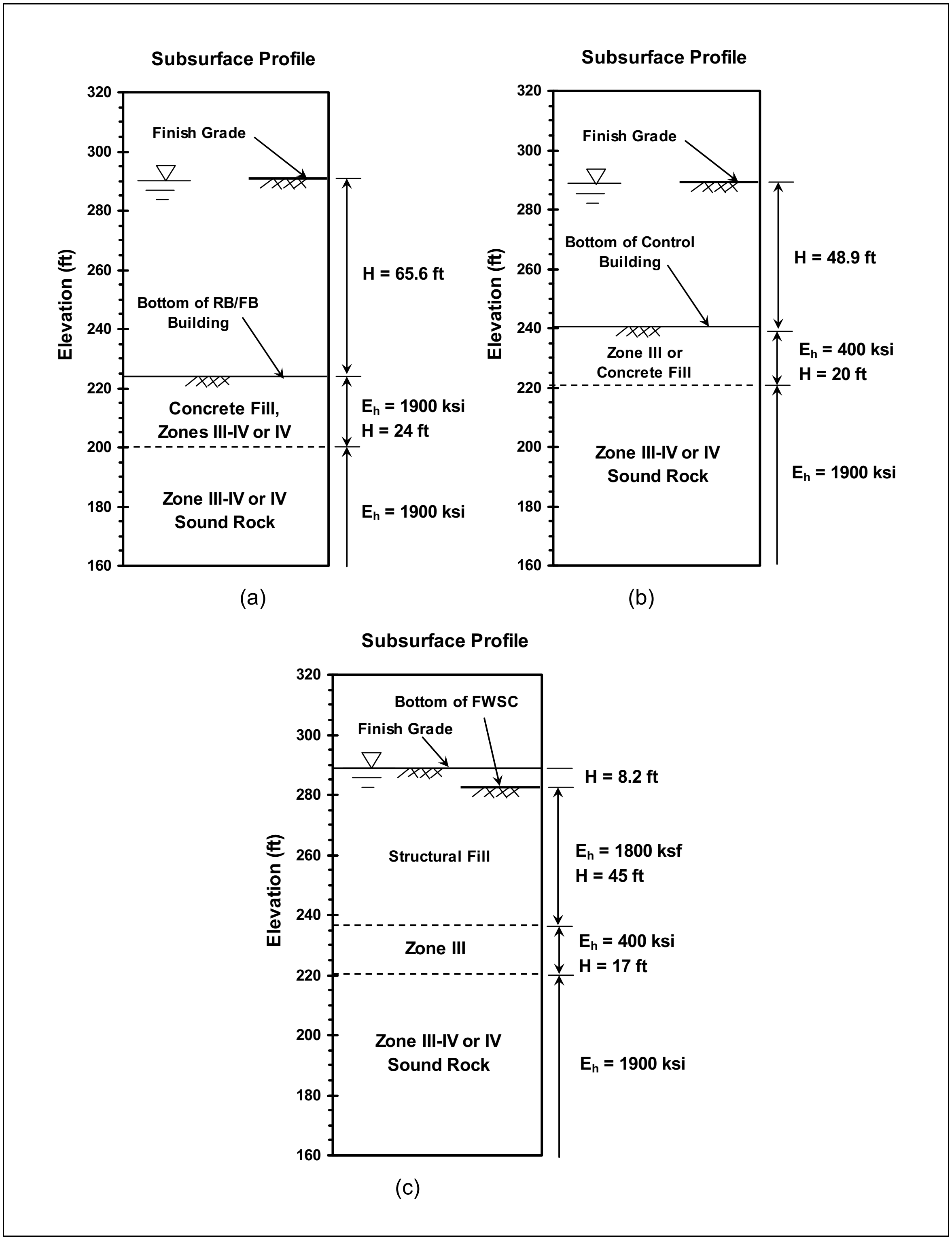
NAPS ESP COL 2.5-5 **Figure 2.5-249 Maximum Acceleration versus Depth, Natural Soil Profile, Low Frequency Input**



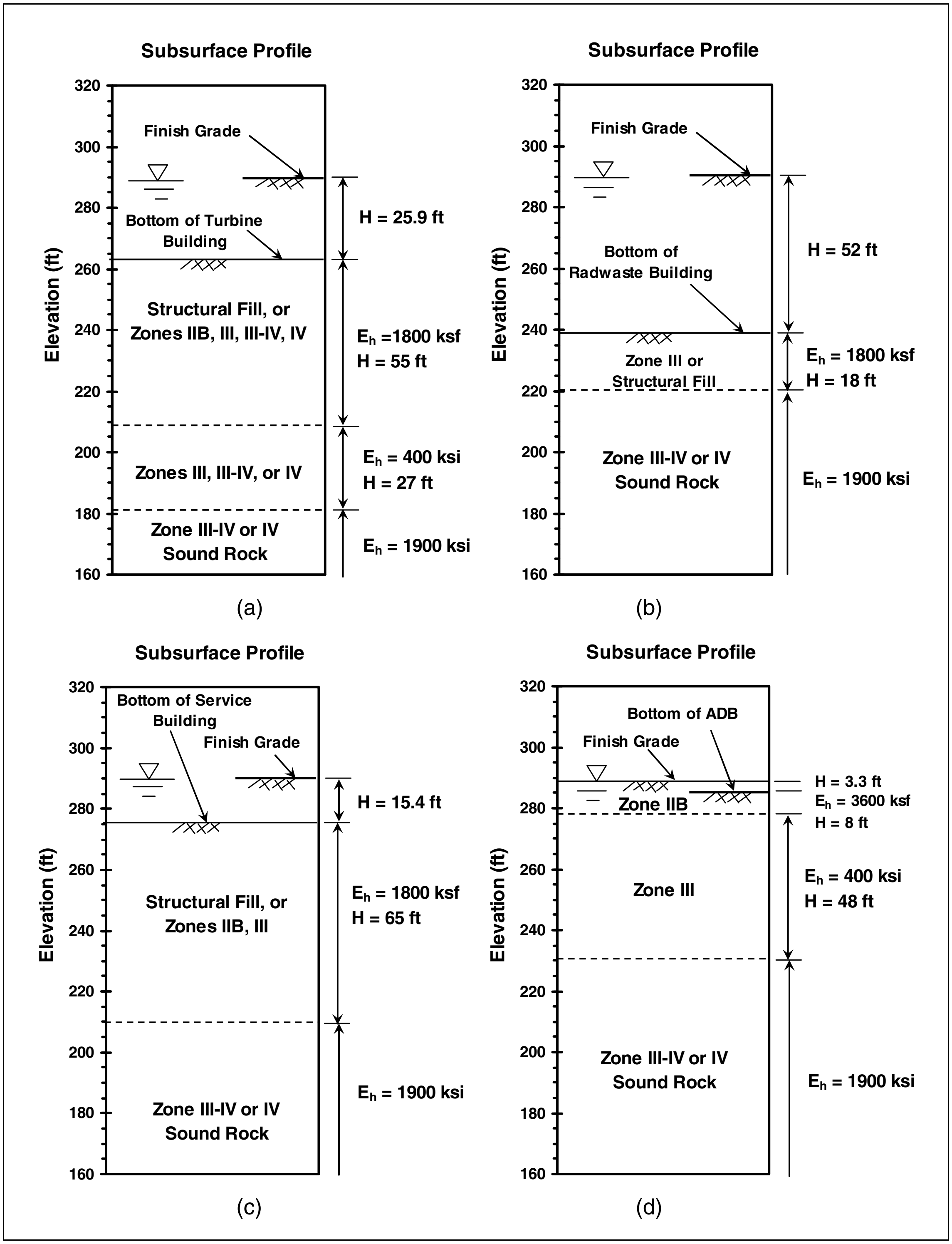
NAPS ESP COL 2.5-5 **Figure 2.5-250 Maximum Acceleration versus Depth, Natural Soil Profile, High Frequency Input**



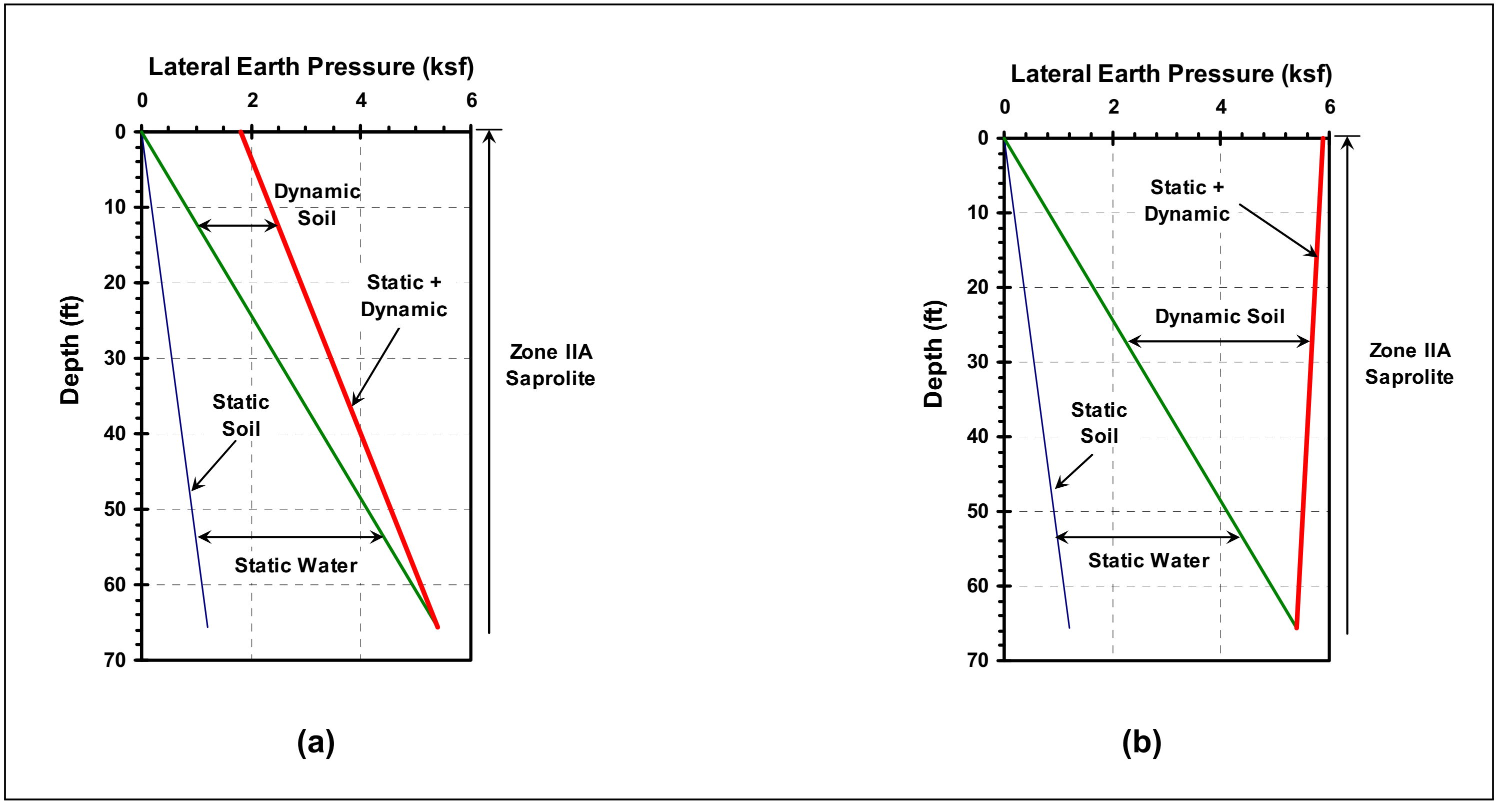
NAPS ESP COL 2.5-6 **Figure 2.5-251 Subsurface Profiles Below the Seismic Category I Structures: (a) Reactor/Fuel Building; (b) Control Building; (c) FWSC**



NAPS ESP COL 2.5-6 **Figure 2.5-252 Subsurface Profiles below the non-Seismic Category I Structures: (a) Turbine Building; (b) Radwaste Building; (c) Service Building; (d) Ancillary Diesel Building**



NAPS ESP COL 2.5-6 **Figure 2.5-253 Active Earth Pressure on Yielding Walls: (a) From In-Situ Zone IIA Saprolite - Peak Ground Acceleration, a_{max} = 0.31g; (b) From In-Situ Zone IIA Saprolite - Peak Ground Acceleration, a_{max} = 0.56g**



NAPS ESP COL 2.5-6 **Figure 2.5-254 Lateral Earth Pressure on Permanent Non-Yielding Walls (Reactor Building Case)**

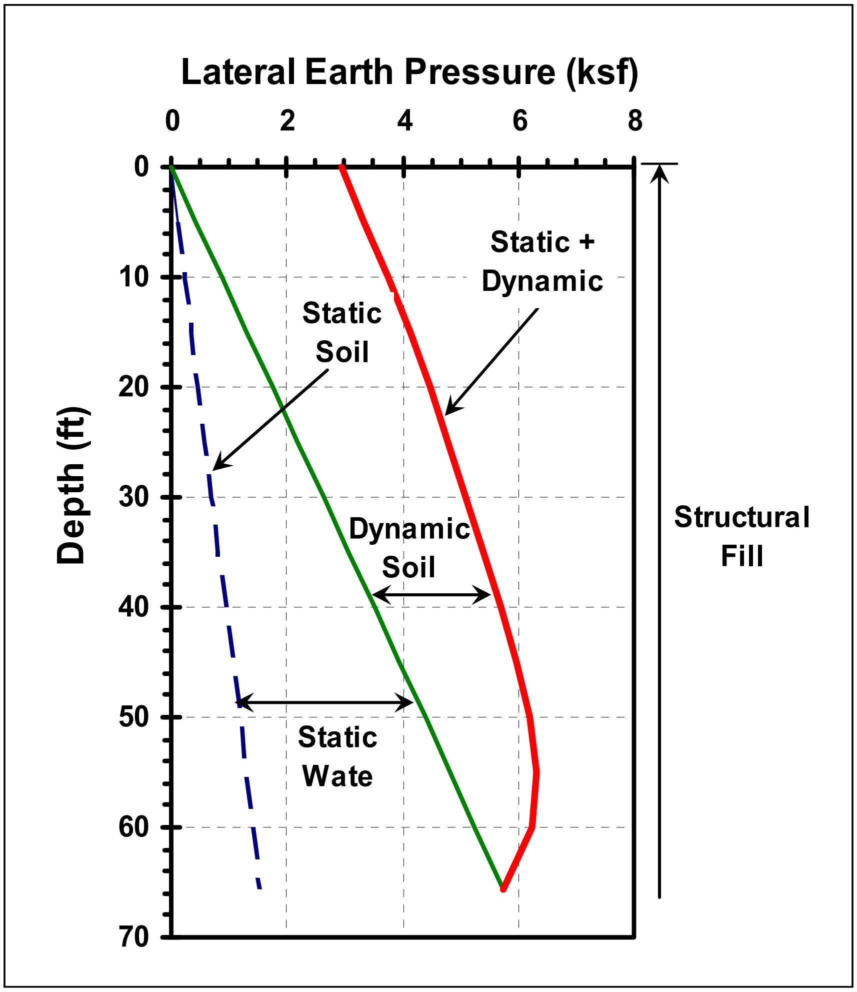
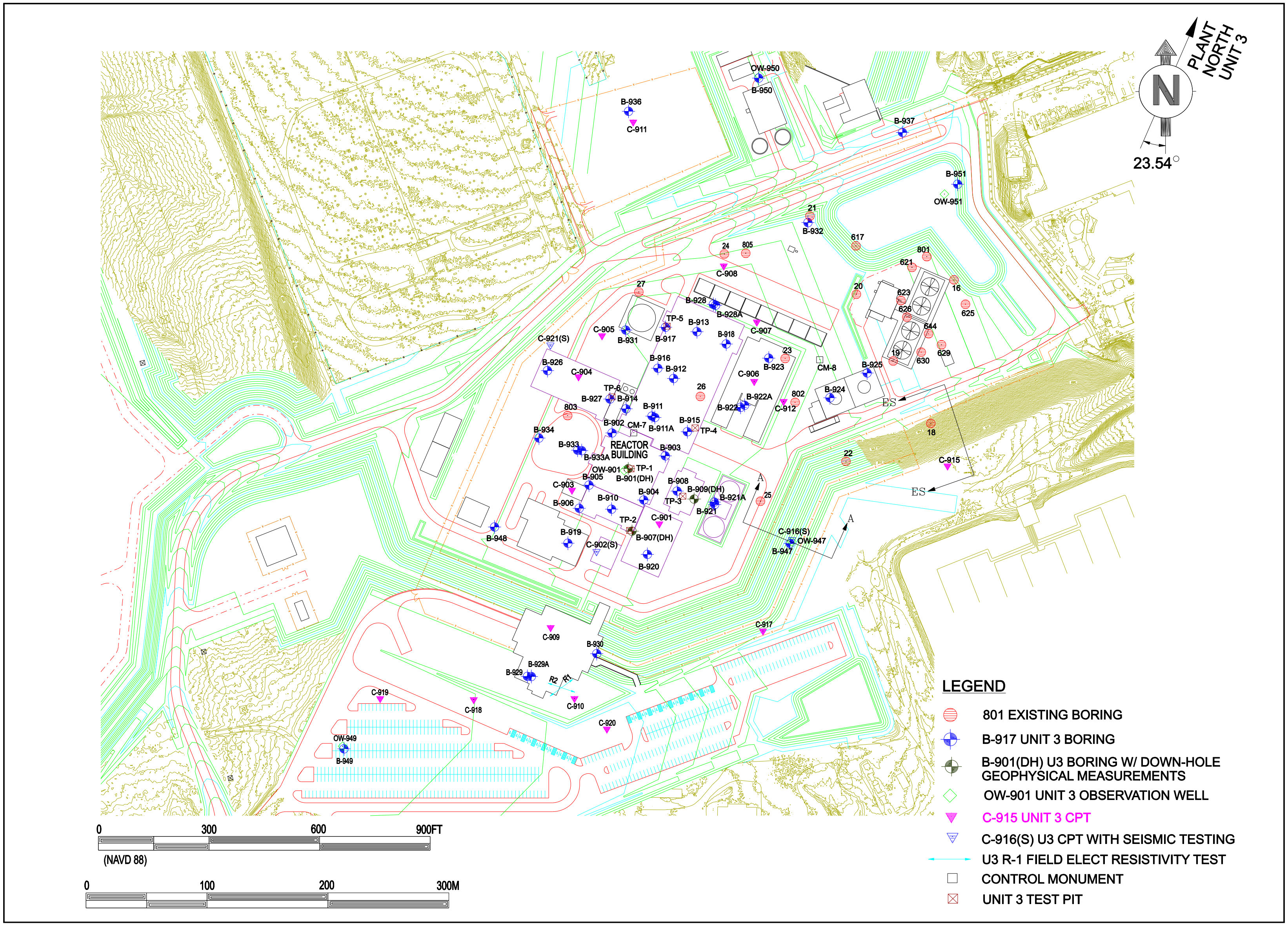


Figure 2.5-255 Grading Plan with Boring Locations



NAPS COL 2.0-30-A **Figure 2.5-256 Cross-Section of Existing Slope (ES)**

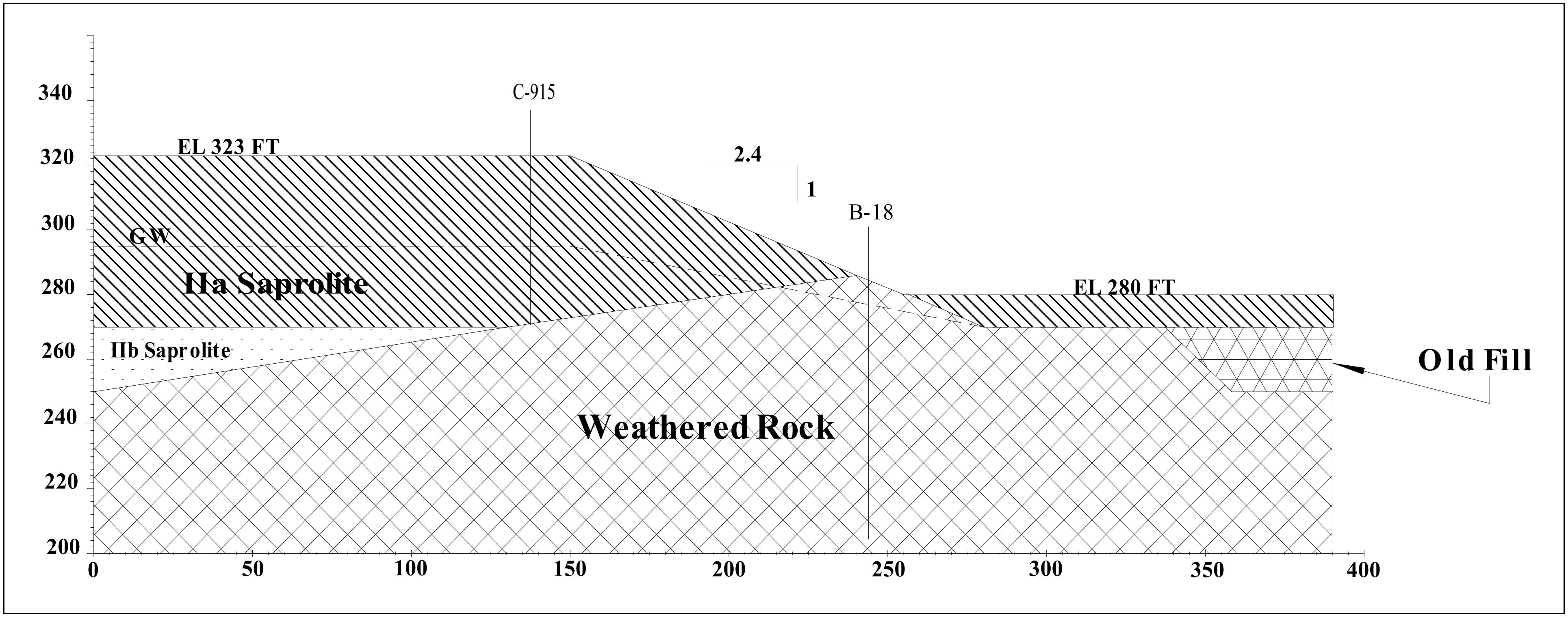
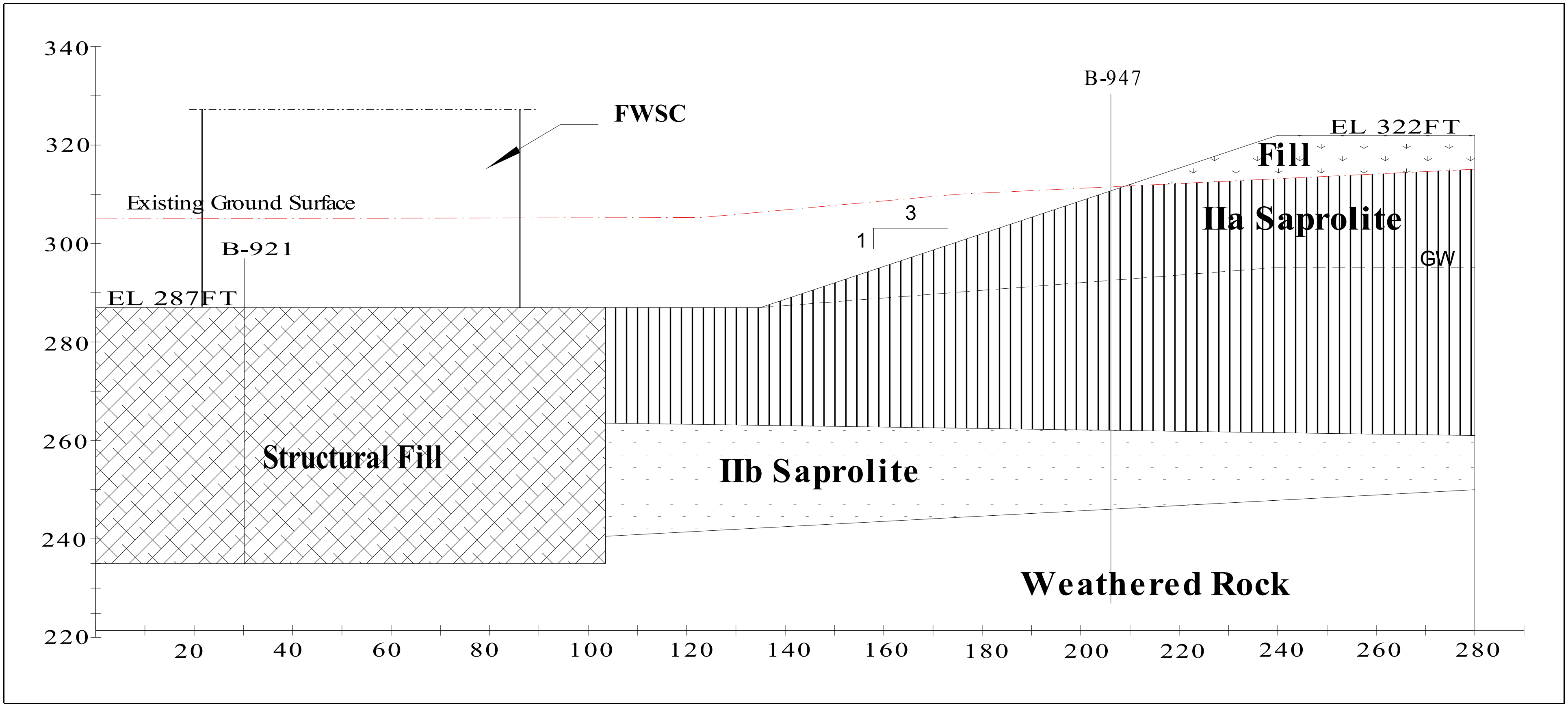
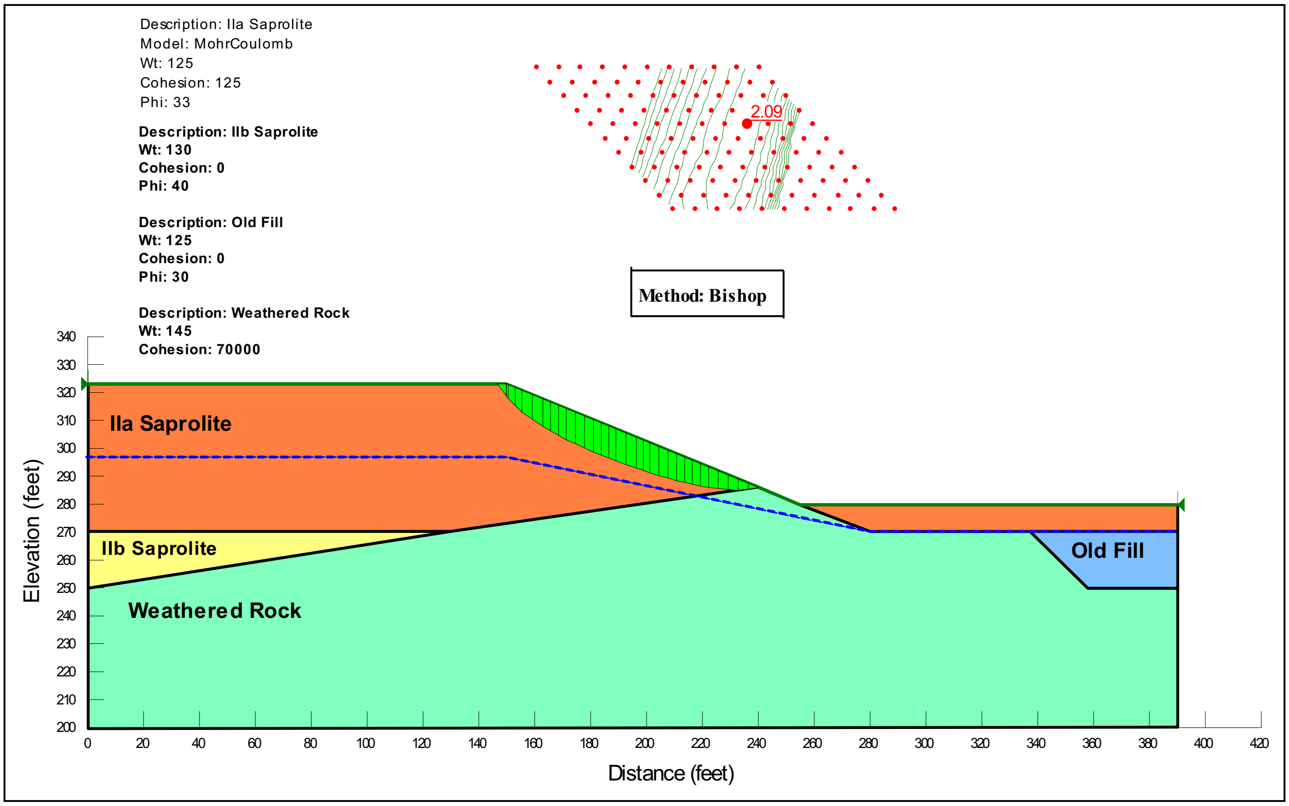


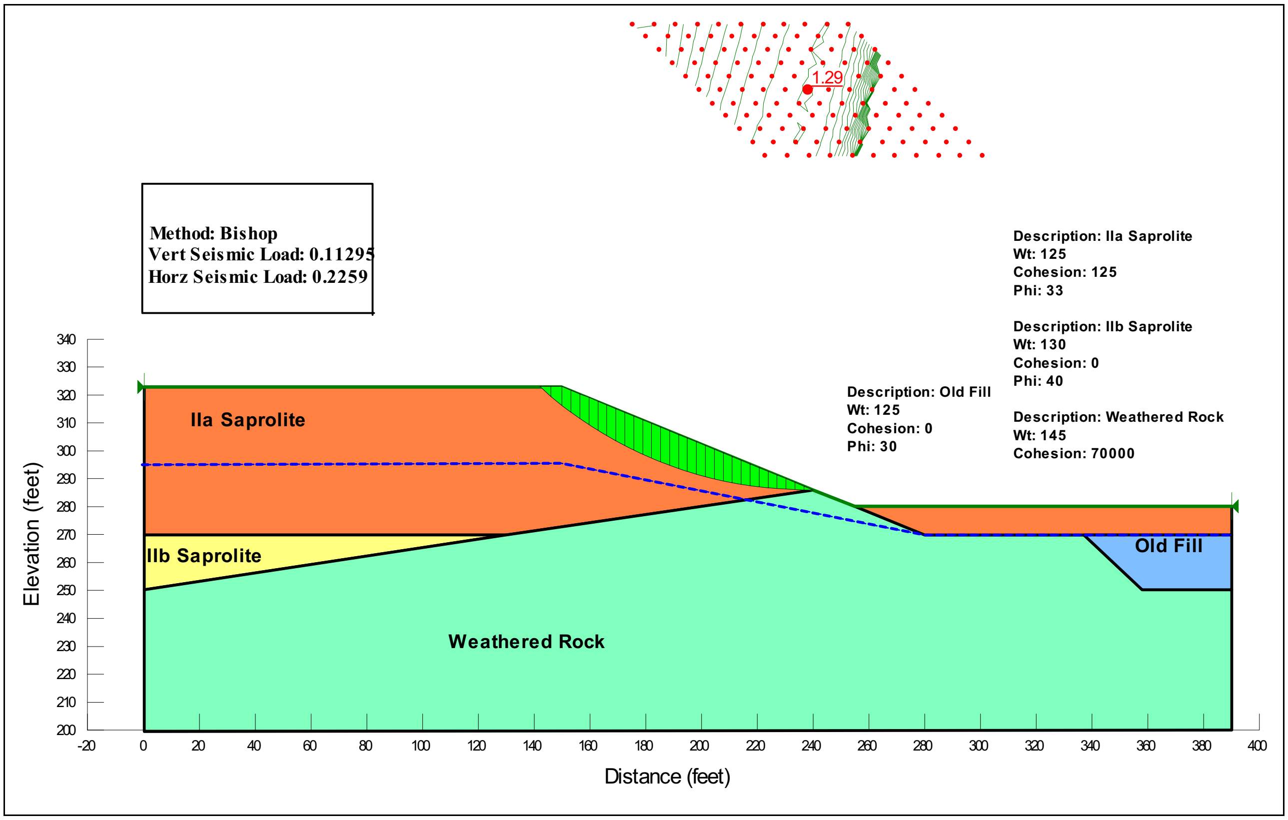
 Figure 2.5-257 Cross-Section of New Slope (A-A)



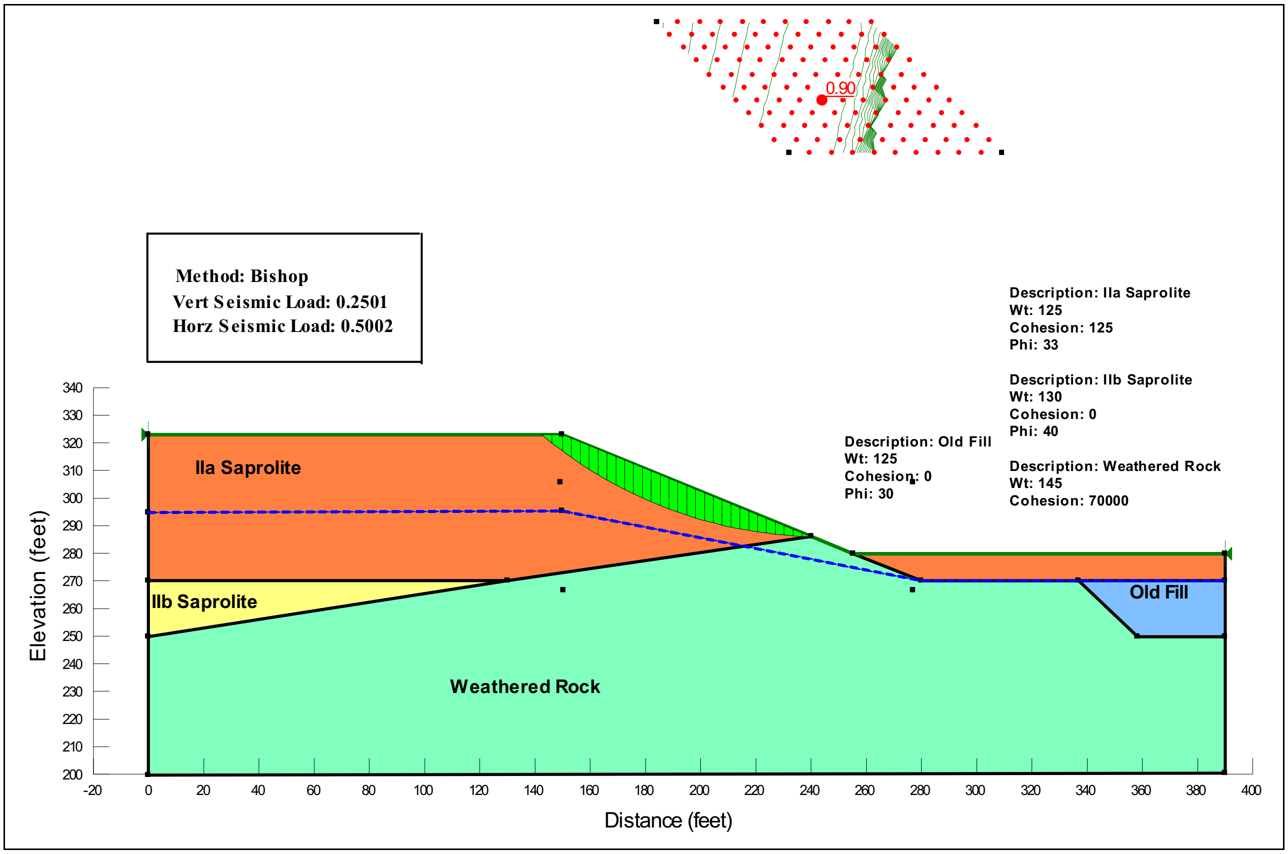
NAPS COL 2.0-30-A **Figure 2.5-258 Slope Stability Analysis; Existing Slope; Long-Term**



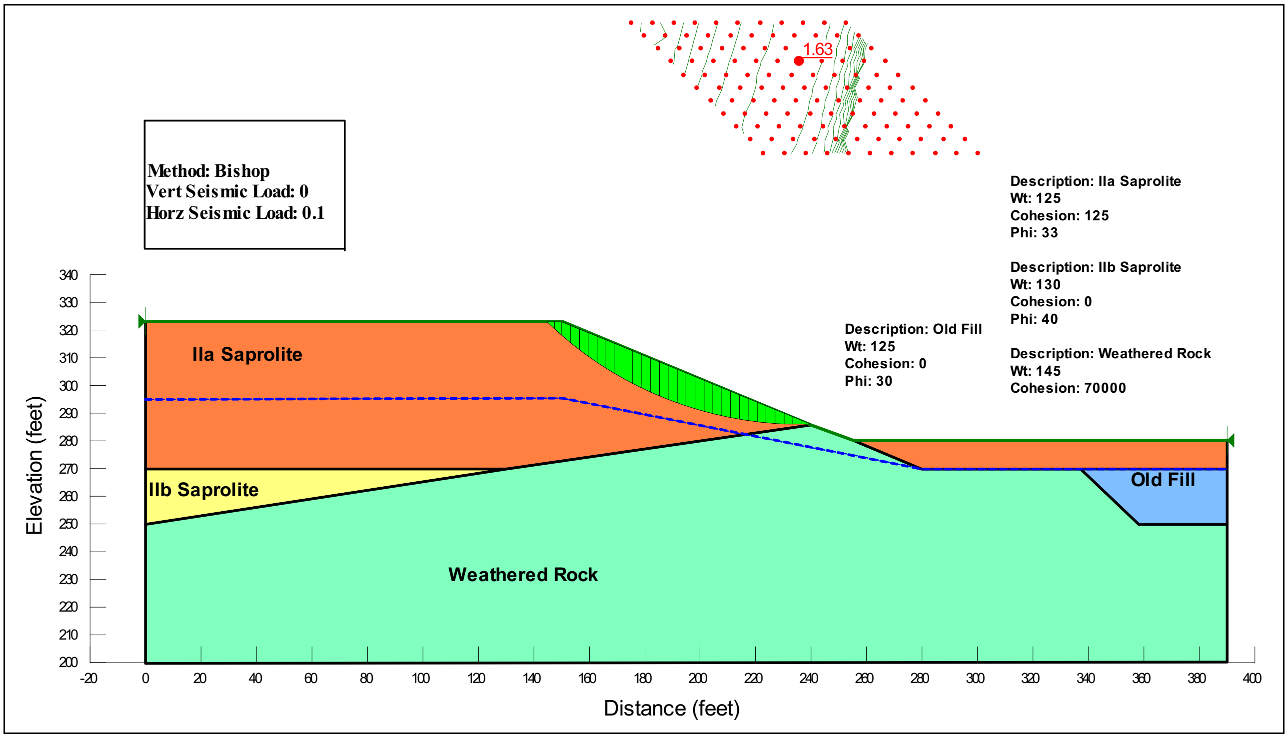
NAPS ESP COL 2.5-10 **Figure 2.5-259 Slope Stability Analysis; Existing Slope; Pseudo-Static; LF**



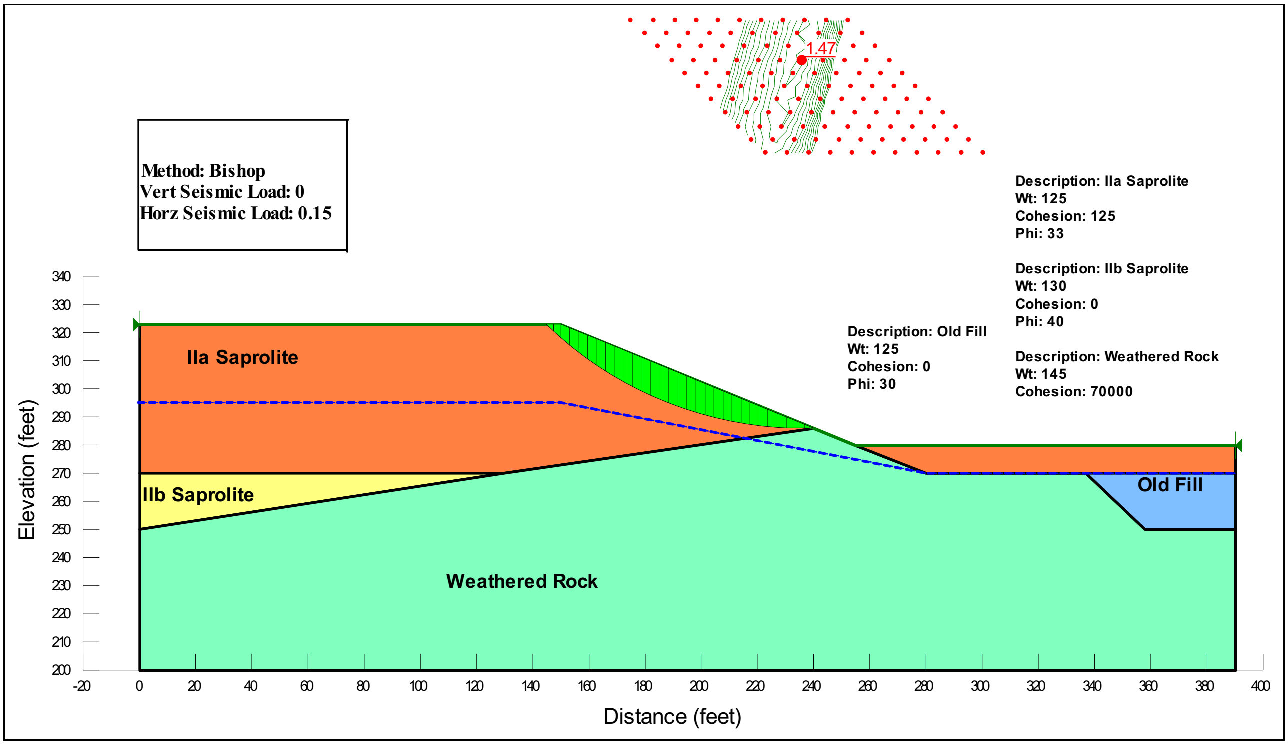
NAPS ESP COL 2.5-10 **Figure 2.5-260 Slope Stability Analysis; Existing Slope; Pseudo-Static; HF**



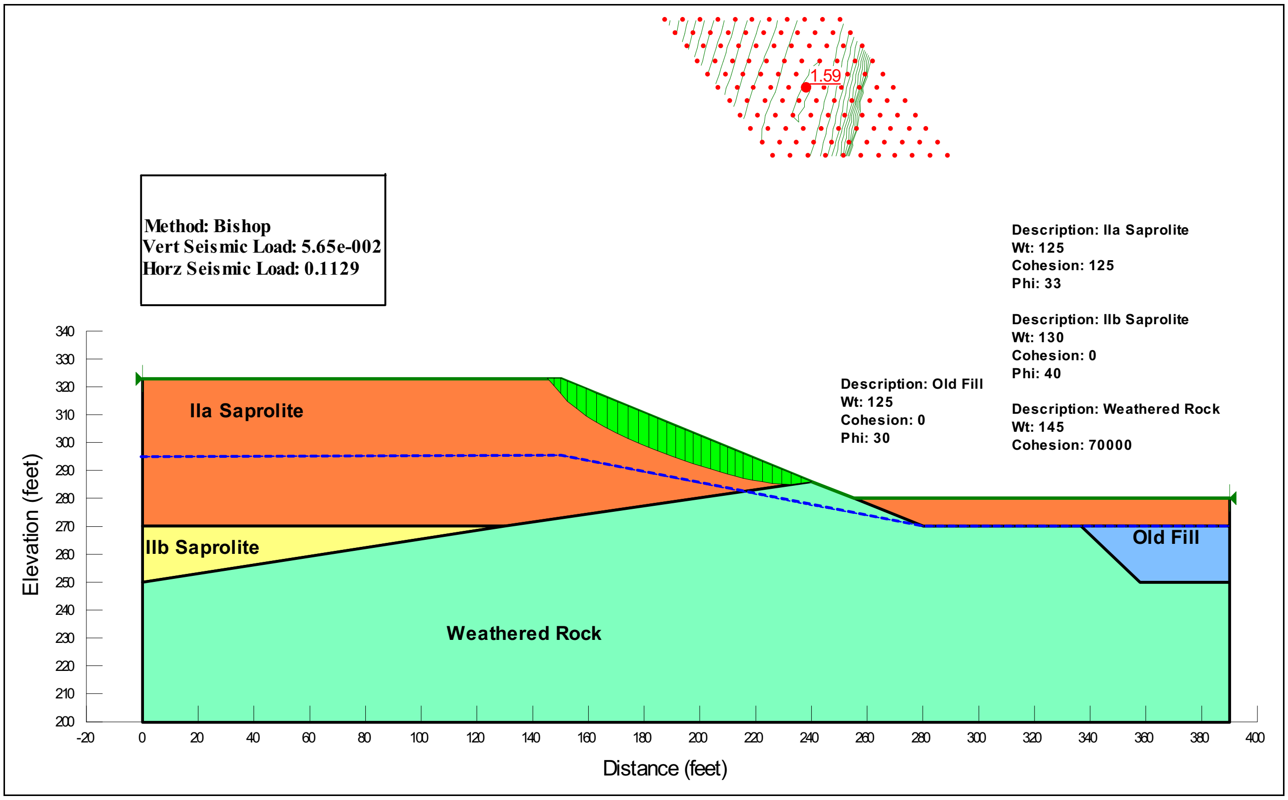
NAPS ESP COL 2.5-10 **Figure 2.5-261 Slope Stability Analysis; Existing Slope; Seed Approach; Acceleration of 0.1g**



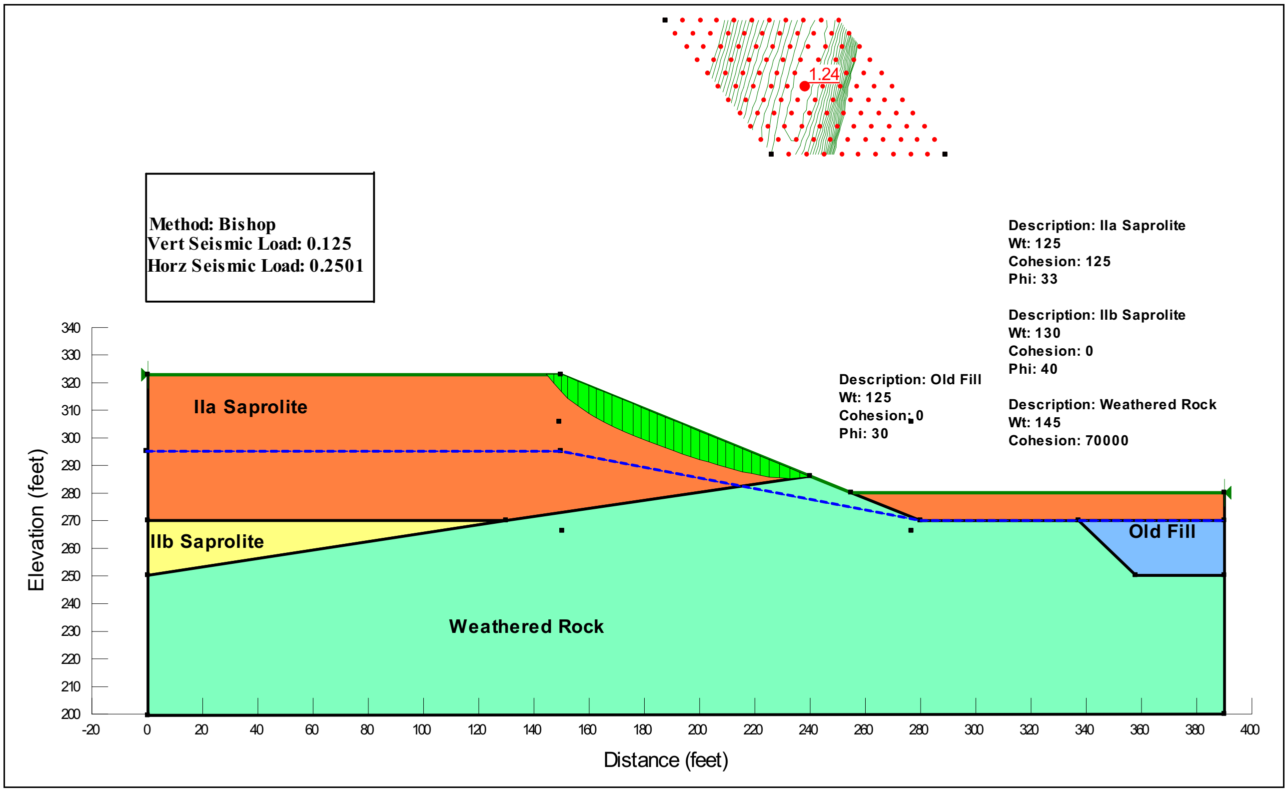
NAPS ESP COL 2.5-10 **Figure 2.5-262 Slope Stability Analysis; Existing Slope; Seed Approach; Acceleration of 0.15g**



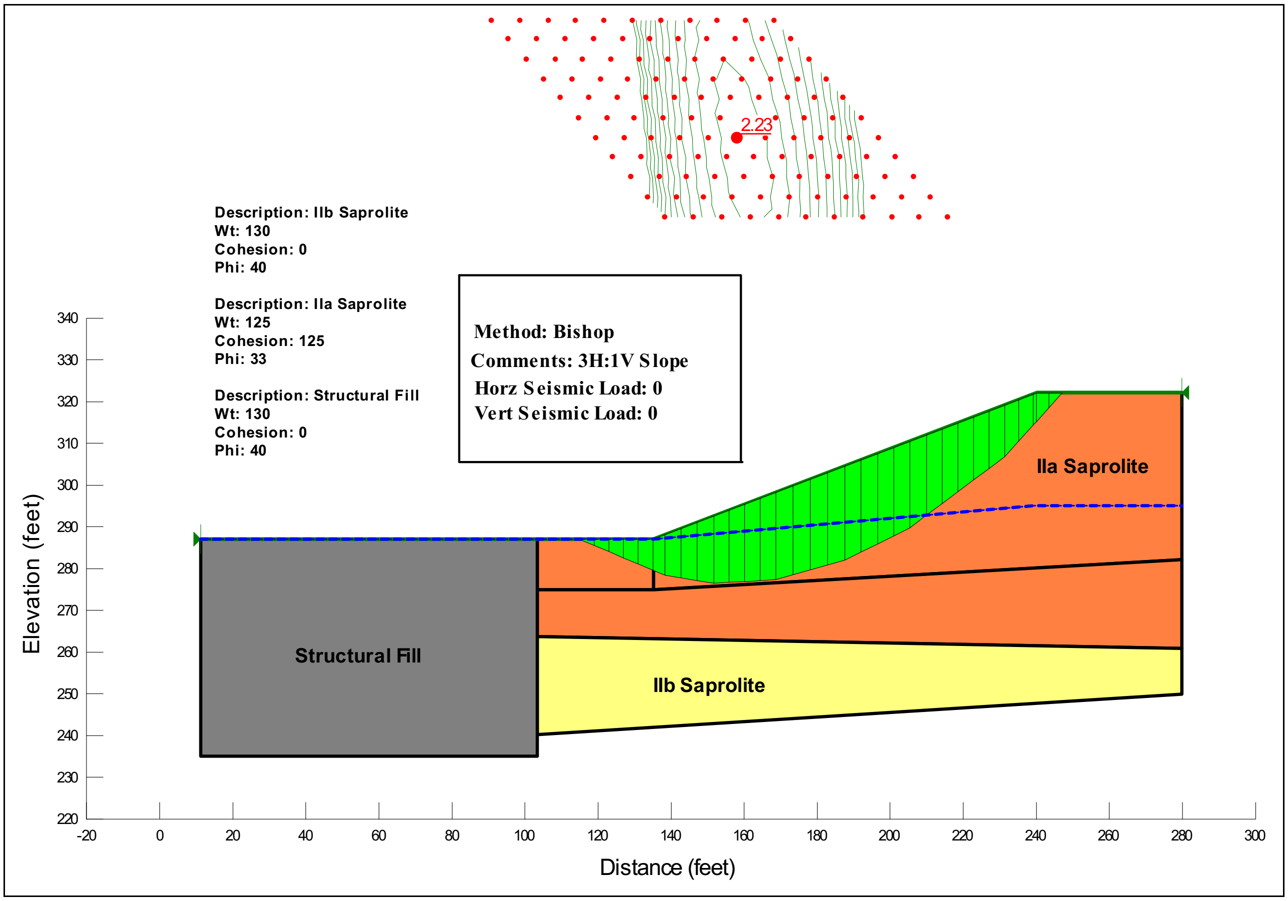
NAPS ESP COL 2.5-10 **Figure 2.5-263 Slope Stability Analysis; Existing Slope; Kramer Approach; LF**



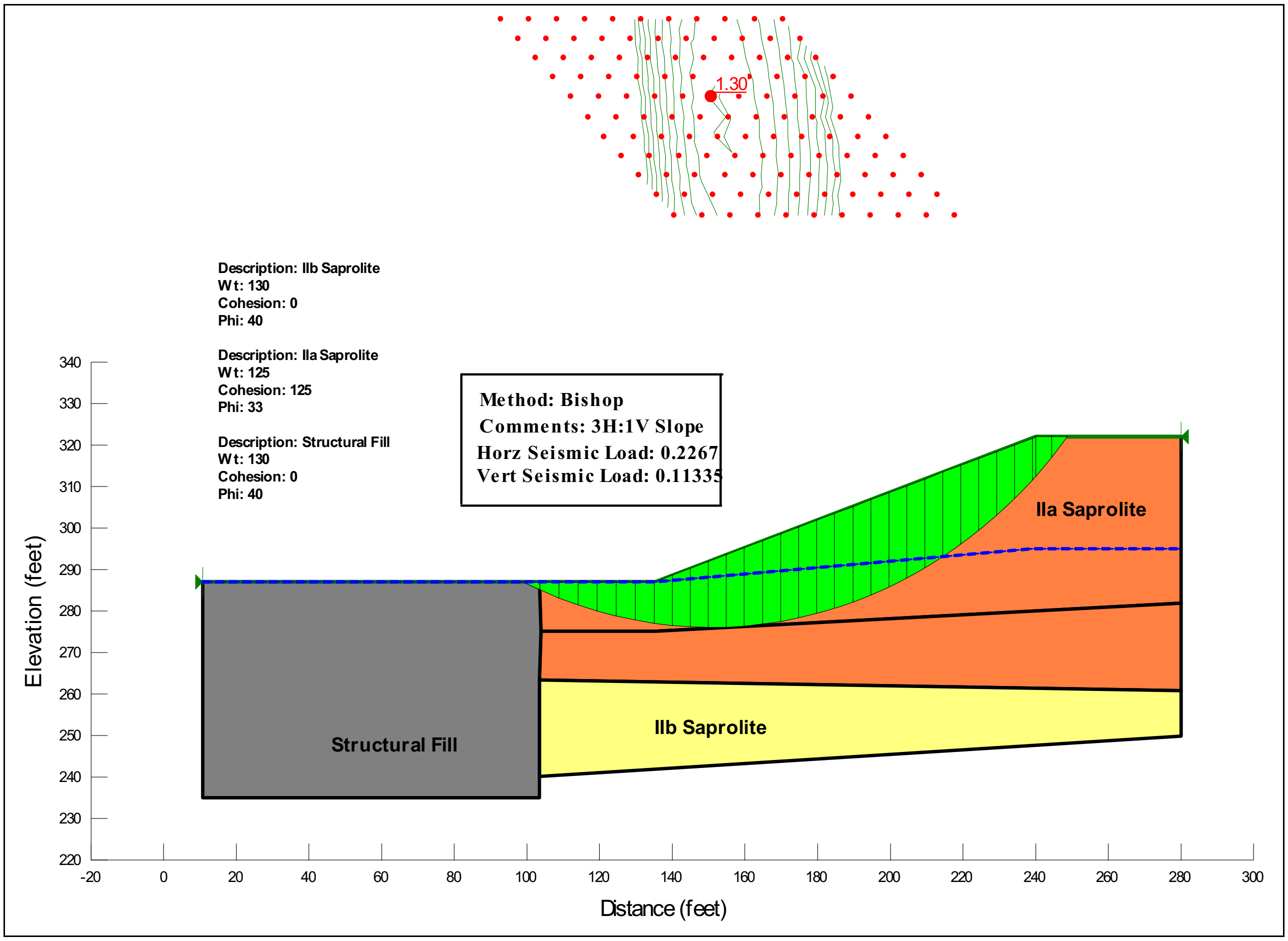
NAPS ESP COL 2.5-10 **Figure 2.5-264 Slope Stability Analysis; Existing Slope; Kramer Approach; HF**



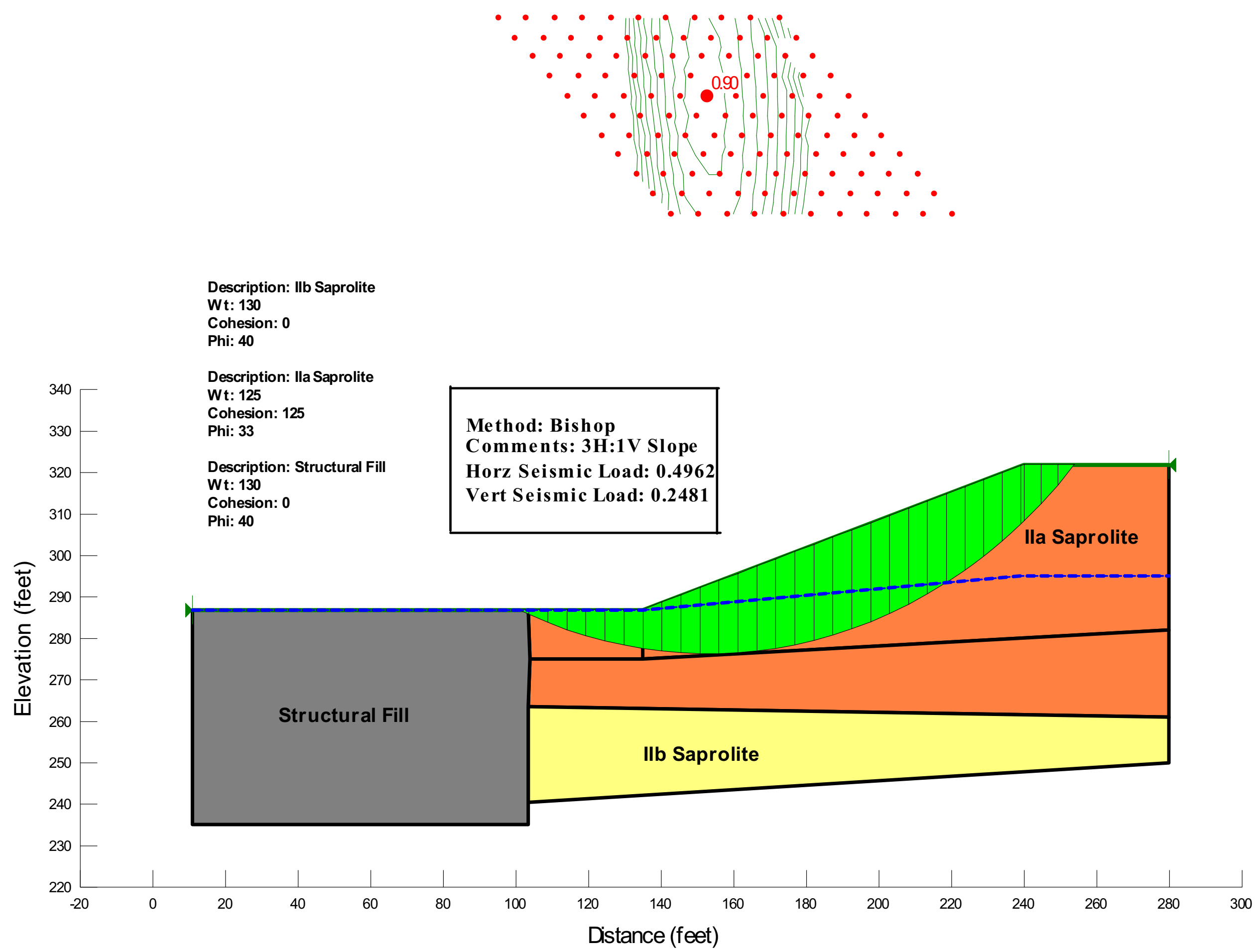
NAPS COL 2.0-30-A **Figure 2.5-265 Slope Stability Analysis; New Slope; Long-Term**



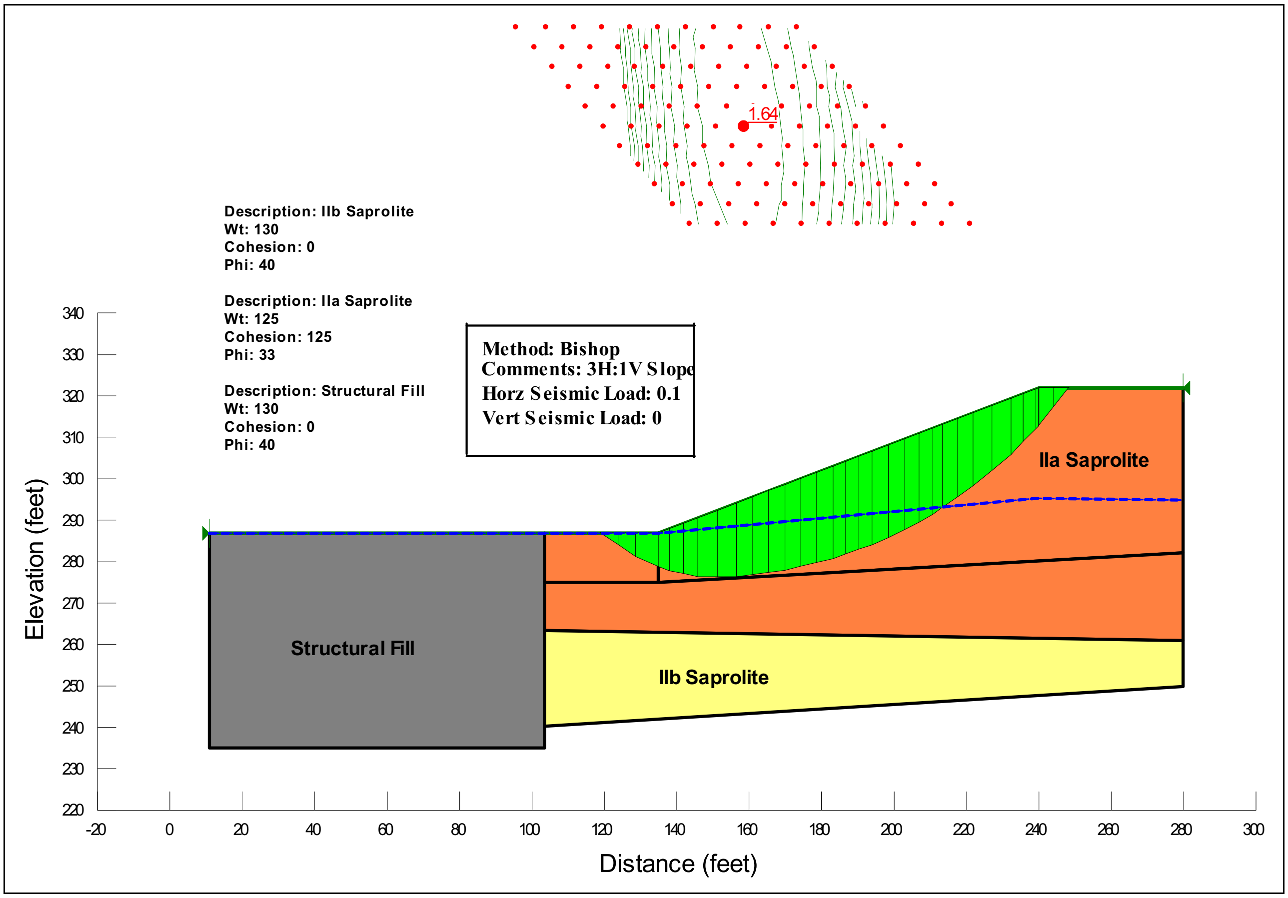
NAPS ESP COL 2.5-10 **Figure 2.5-266 Slope Stability Analysis; New Slope; Pseudo-Static; LF**



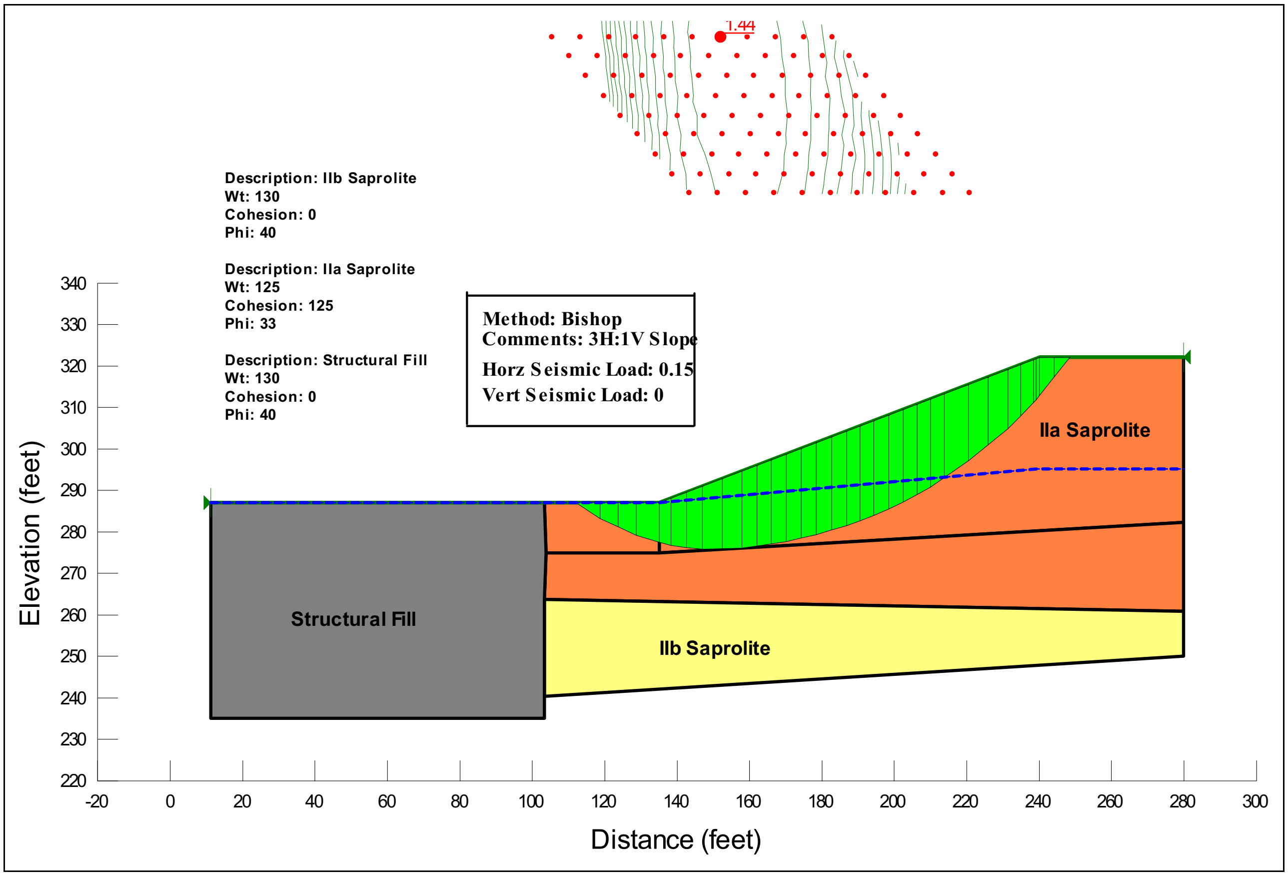
NAPS ESP COL 2.5-10 **Figure 2.5-267 Slope Stability Analysis; New Slope; Pseudo-Static; HF**



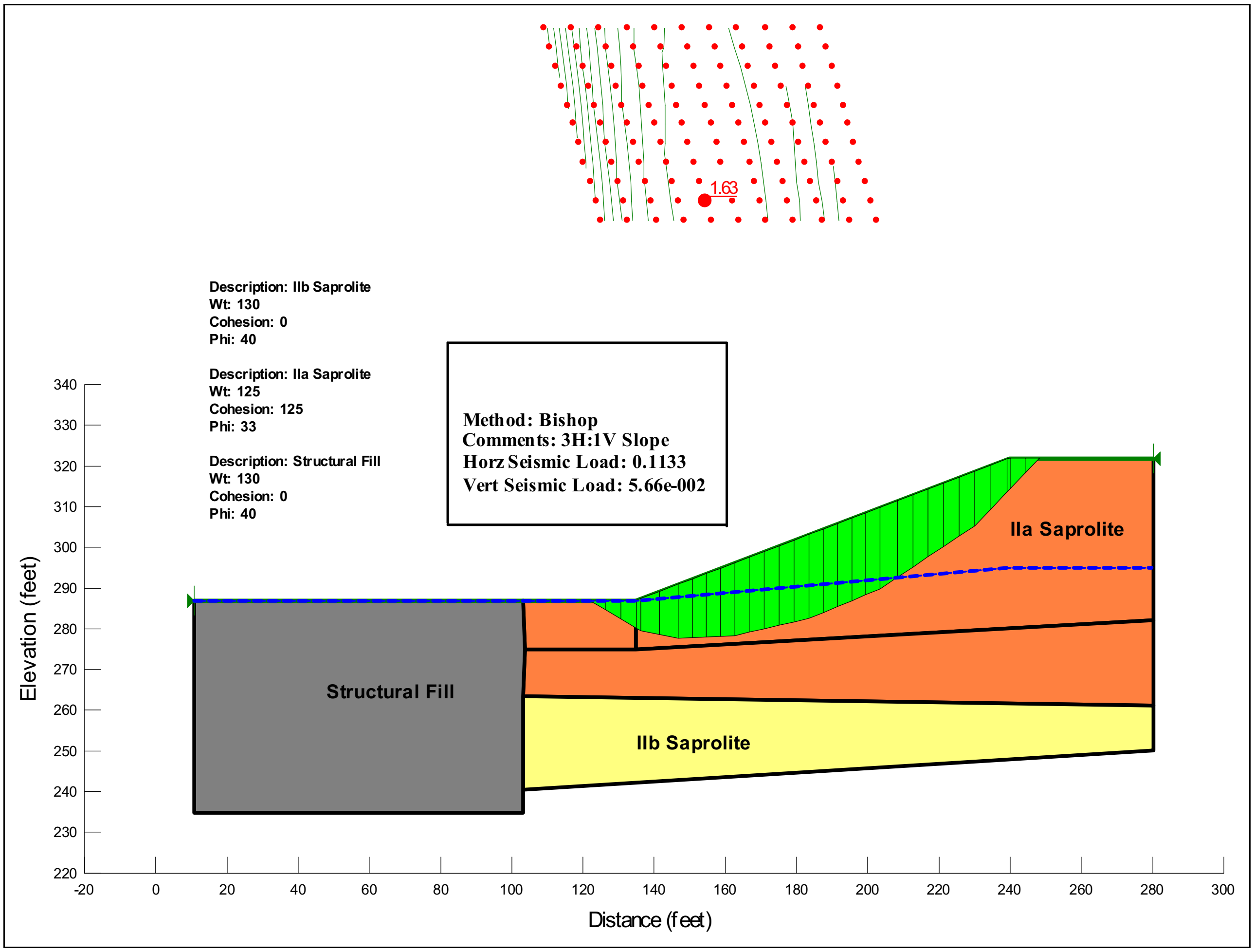
NAPS ESP COL 2.5-10 **Figure 2.5-268 Slope Stability Analysis; New Slope; Seed Approach; Acceleration of 0.1g**



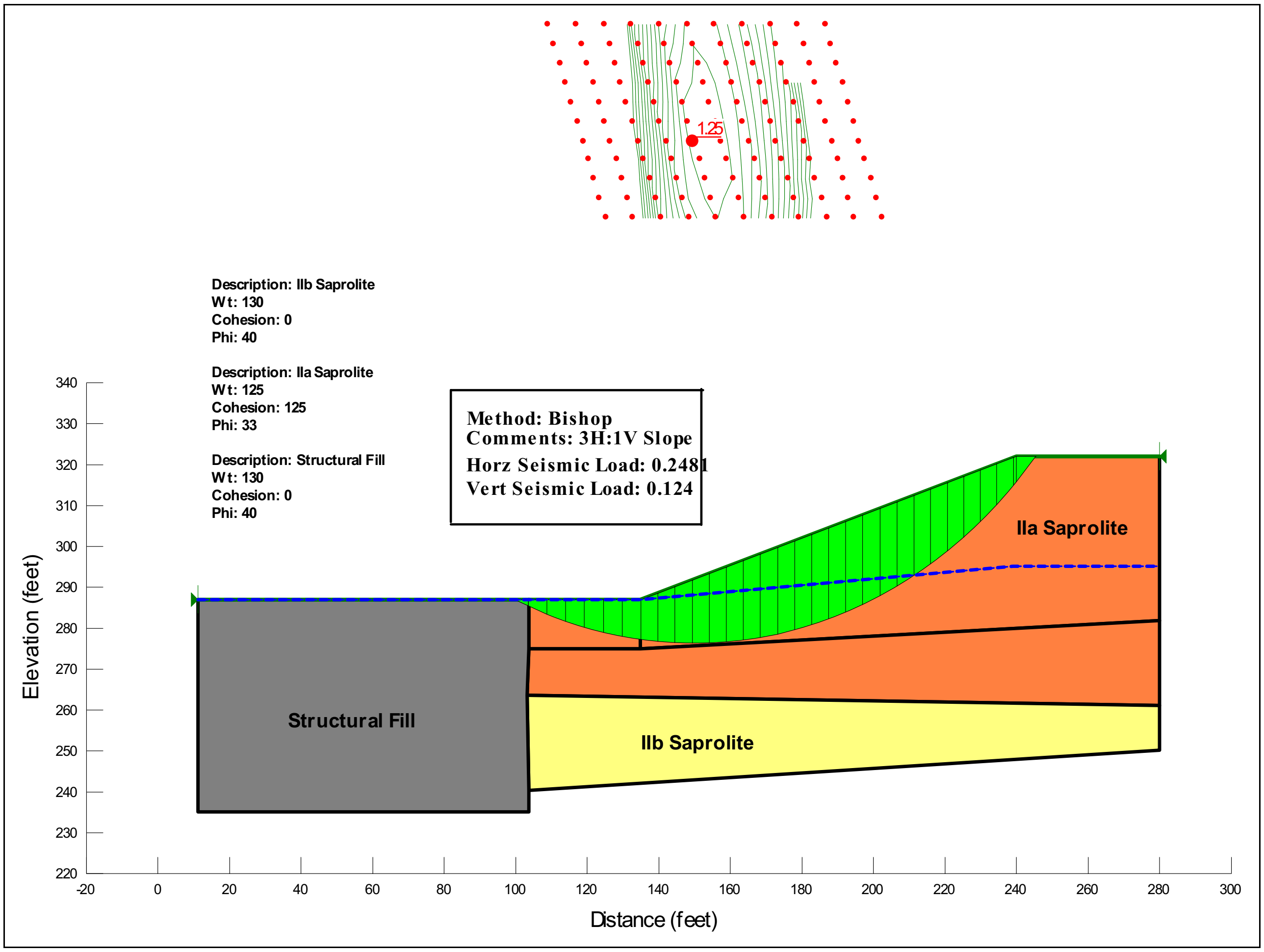
NAPS ESP COL 2.5-10 **Figure 2.5-269 Slope Stability Analysis; New Slope; Seed Approach; Acceleration of 0.15g**



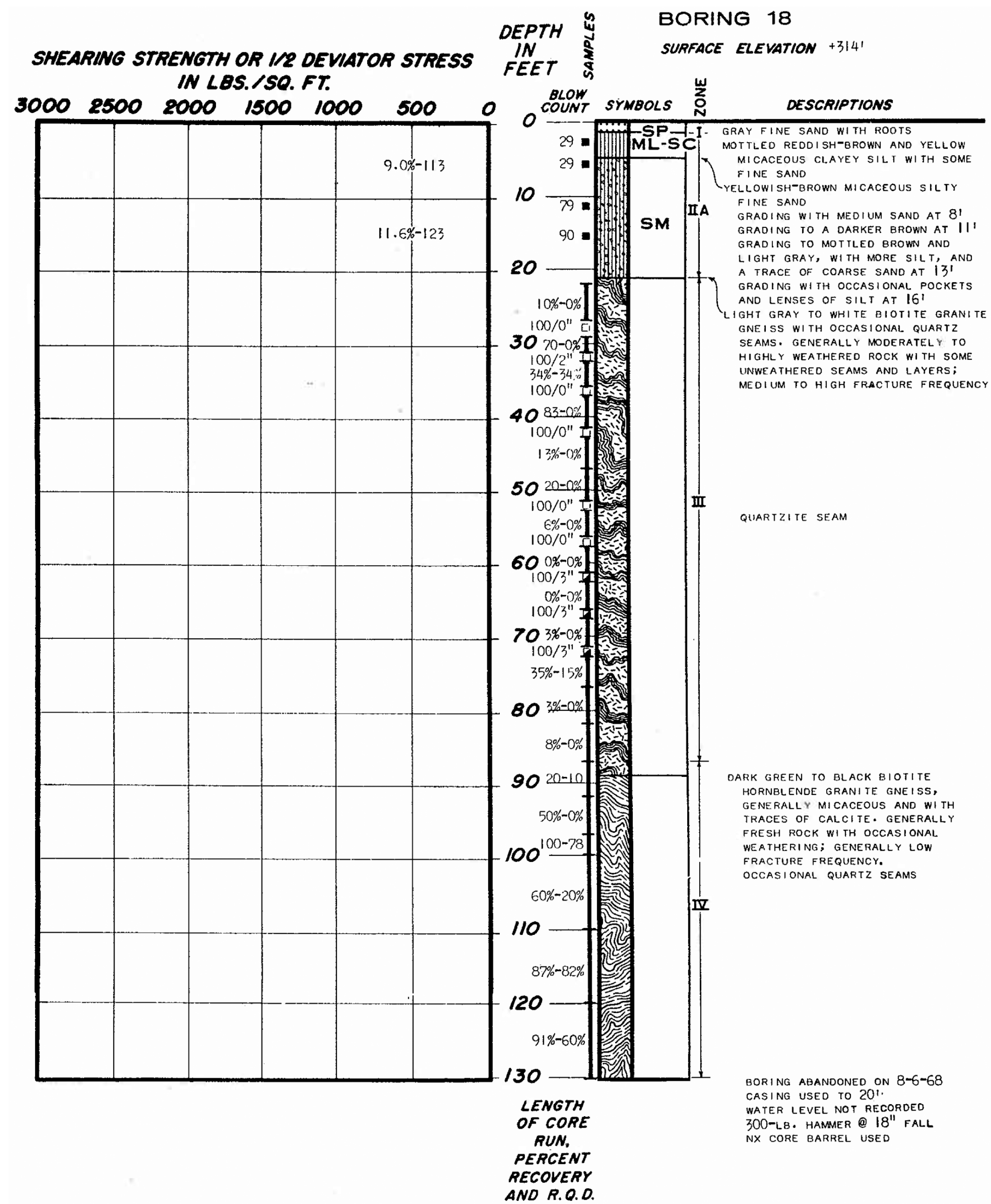
NAPS ESP COL 2.5-10 **Figure 2.5-270 Slope Stability Analysis; New Slope; Kramer Approach; LF**



NAPS ESP COL 2.5-10 **Figure 2.5-271 Slope Stability Analysis; New Slope; Kramer Approach; HF**



NAPS COL 2.0-30-A **Figure 2.5-272 Log of Boring B-18**

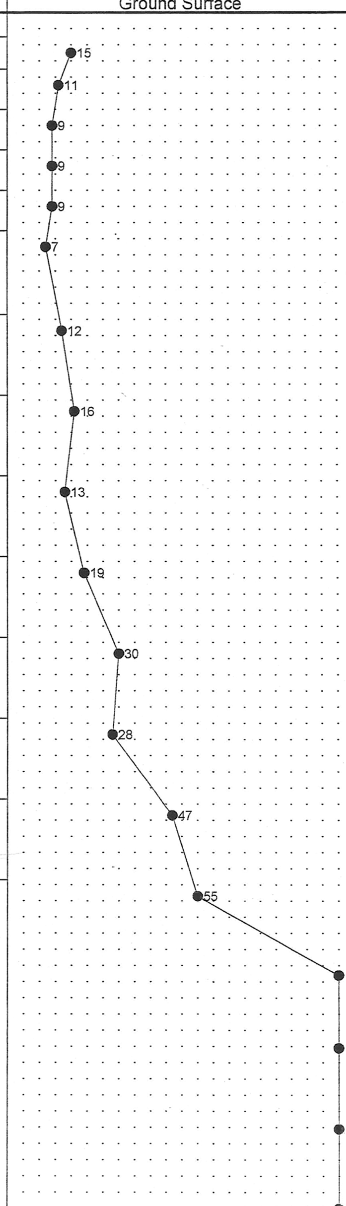


NAPS COL 2.0-30-A

Figure 2.5-273 Log of Boring B-947

MACTEC PROJECT NO.: 6468-06-1472	COUNTY Louisa, VA	GEOLOGIST S.Lehman	
SITE DESCRIPTION NORTH ANNA COL			GROUND WATER (ft)
BORING NO. B-947	DRILL METHOD: Mud Rotary	SAMPLE METHODS: SPT	0 HR. ND
COLLAR ELEV. 312.5 ft (NAVD88)	NORTHING 3,909,575 US ft (NAD83)	EASTING 11,686,367 US ft (NAD83)	24 HR. 19.5
TOTAL DEPTH 88.8 ft	DRILL MACHINE CME 55LC Track	DRILLER: D. White	HAMMER TYPE 140 lbs Auto
DATE STARTED 9/27/06	COMPLETED 9/28/06	CORE BARREL TYPE: NA	

ELEV. (ft)	DEPTH (ft)	BLOW COUNT 0.5ft	0.5ft	0.5ft	BLOWS PER FOOT (0–100)	SAMP. NO.
312.5					Ground Surface	
311.0	1.5	4	6	9	15	947-1
309.0	3.5	6	5	6	11	947-2
306.5	6.0	4	3	6	9	947-3
304.0	8.5	4	4	5	9	947-4
301.5	11.0	2	4	5	9	947-5
299.0	13.5	2	3	4	7	947-6
293.8	18.7	5	6	6	12	947-7
288.8	23.7	5	6	10	16	947-8
283.8	28.7	4	3	10	13	947-9
278.8	33.7	3	7	12	19	947-10
273.8	38.7	12	14	16	30	947-11
268.8	43.7	9	11	17	28	947-12
263.8	48.7	13	23	24	47	947-13
258.8	53.7	23	27	28	55	947-14
253.8	58.7	36	50/0.4			947-15
248.8	63.7	50/0.4				947-16
243.8	68.7	50/0.4				947-17
238.8	73.7	50/0.3				947-18



SOIL AND ROCK DESCRIPTION

312.5 — 0.0
Elastic SILT (MH) light gray (2.5Y 7/2) and light greenish gray (GLEY1 8/5GY), dry to moist, stiff (Residual Soil)

304.5 — 8.0
Silty, SAND (SM) strong brown (7.5YR 4/6), moist, loose (Residual Soil)

299.5 — 13.0
Sandy SILT (ML) light gray (2.5Y 7/2), moist, medium stiff (Residual Soil)

295.5 — 17.0
Silty, fine to medium SAND (SM) reddish yellow (7.5YR 6/6), yellowish brown (2.5 Y 6/4), white (2.5Y 8/1), light yellow (2.5Y 7/3), and light olive brown (2.5Y 5/4&5/3), wet, medium dense to very dense, relict rock fabric (Residual Soil)

254.5 — 58.0
Weathered Rock: Light yellowish brown (2.5Y 6/4) and light brownish gray (2.5Y 6/2), moist, very dense, severely weathered, soft, QUARTZ GNEISS

NORTH ANNA COL ...RE NORTH ANNA COL DATA REPORT REV0.GPJ NORTH ANNA COL.GDT 1/19/07

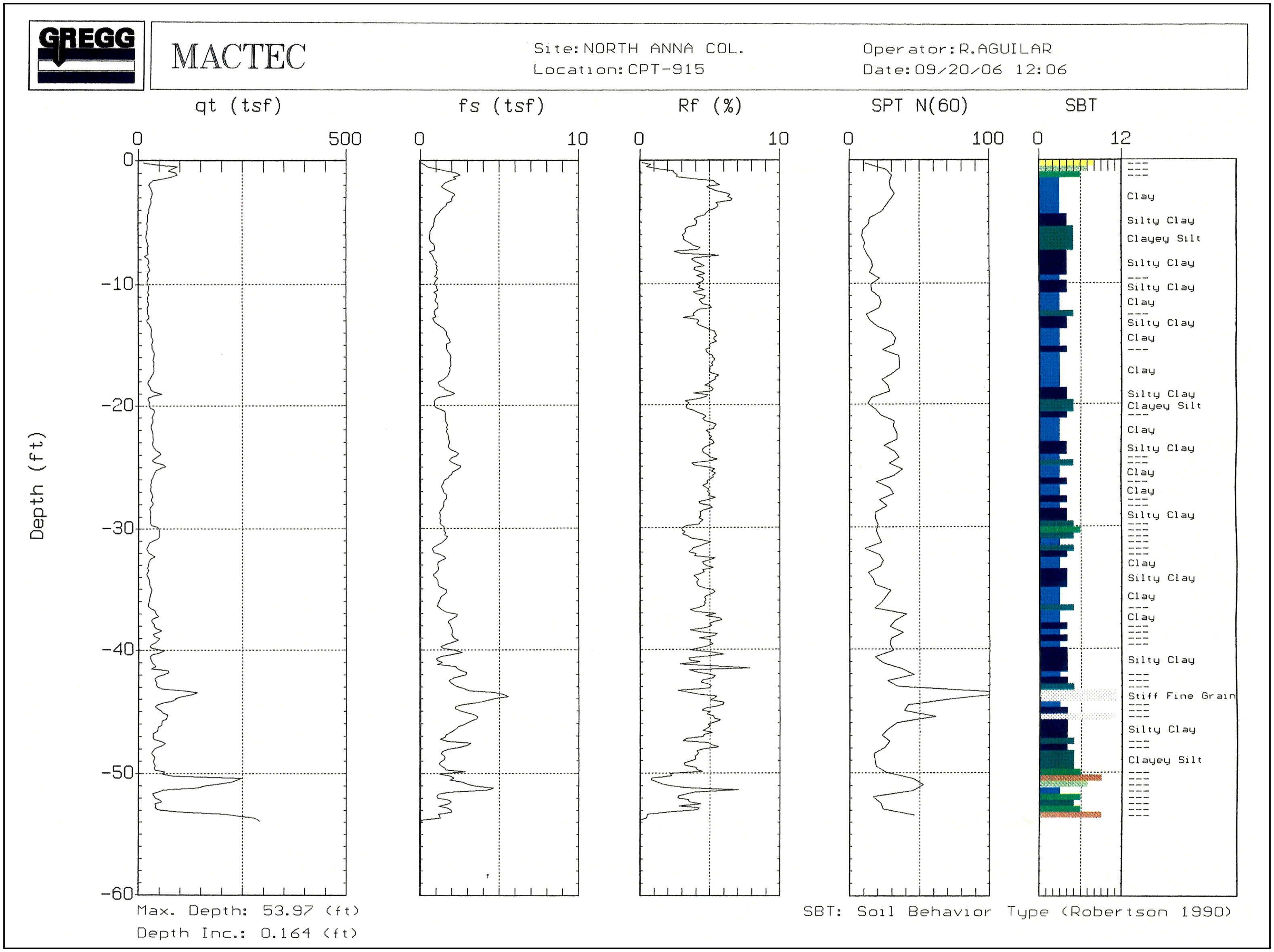
NAPS COL 2.0-30-A **Figure 2.5-273 Log of Boring B-947** *(continued)*

MACTEC PROJECT NO.: 6468-06-1472	COUNTY Louisa, VA	GEOLOGIST S.Lehman	
SITE DESCRIPTION NORTH ANNA COL			GROUND WATER (ft)
BORING NO. B-947	DRILL METHOD: Mud Rotary	SAMPLE METHODS: SPT	0 HR. ND
COLLAR ELEV. 312.5 ft (NAVD88)	NORTHING 3,909,575 US ft (NAD83)	EASTING 11,686,367 US ft (NAD83)	24 HR. 19.5
TOTAL DEPTH 88.8 ft	DRILL MACHINE CME 55LC Track	DRILLER: D. White	HAMMER TYPE 140 lbs Auto
DATE STARTED 9/27/06	COMPLETED 9/28/06	CORE BARREL TYPE: NA	

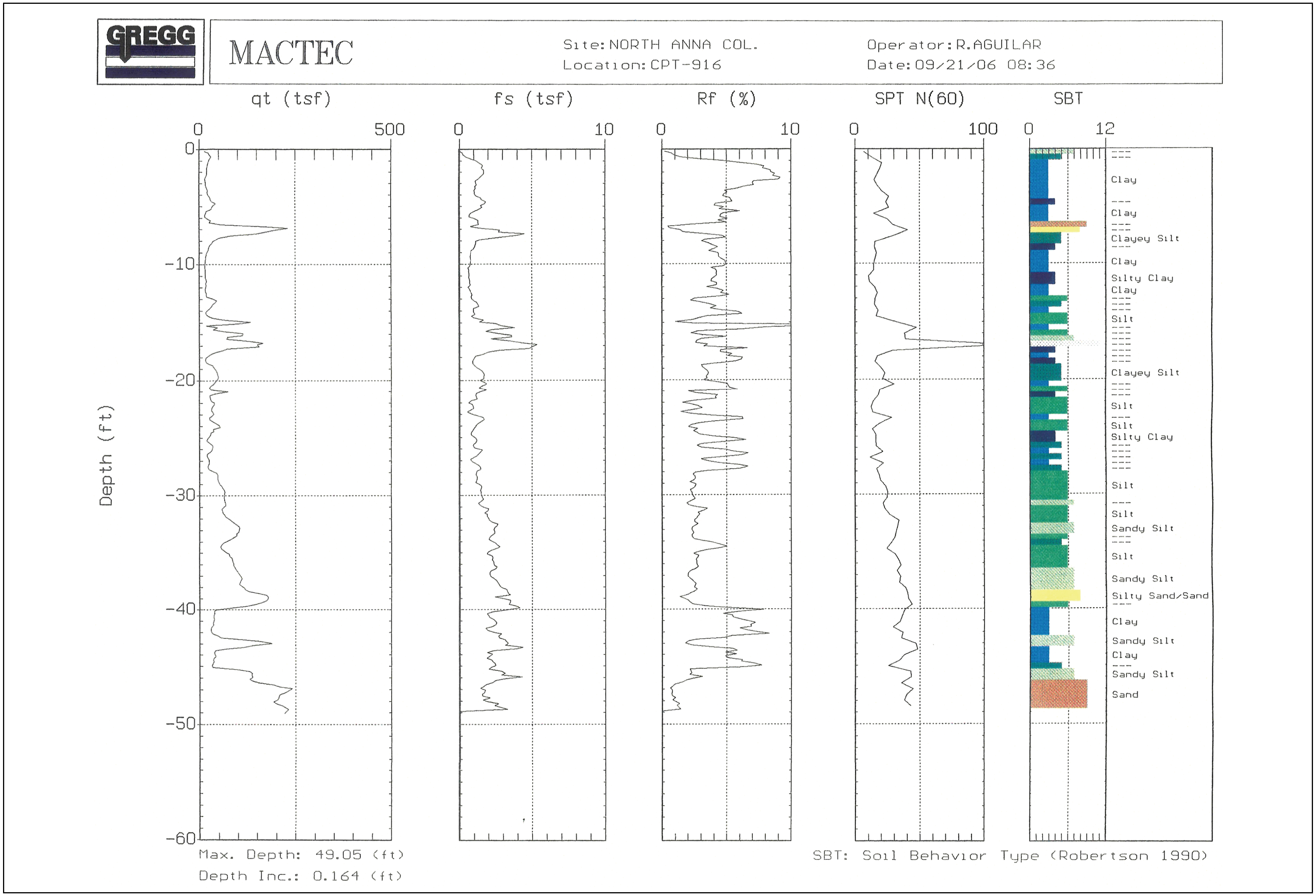
ELEV. (ft)	DEPTH (ft)	BLOW COUNT 0.5ft	0.5ft	0.5ft	BLOWS PER FOOT 0 20 40 60 80 100	SAMP. NO.	MOI	LOG	SOIL AND ROCK DESCRIPTION
237.7					Continued from previous page				
									Weathered Rock: Light yellowish brown (2.5Y 6/4) and light brownish gray (2.5Y 6/2), moist, very dense, severely weathered, soft, QUARTZ GNEISS *(continued)*
233.8	78.7	50/0.2				947-19			
228.8	83.7	50/0.1				947-20			
223.8	88.7	50/0.1				947-21			223.7 — 88.8
									Boring terminated at 88.8 ft in Weathered Rock: Very dense, severely weathered, soft, QUARTZ GNEISS

NORTH ANNA COL CORE NORTH ANNA COL DATA REPORT REV0.GPJ NORTH ANNA COL.GDT 1/19/07

Figure 2.5-274 Log of CPT C-915



NAPS COL 2.0-30-A **Figure 2.5-275 Log of CPT C-916**

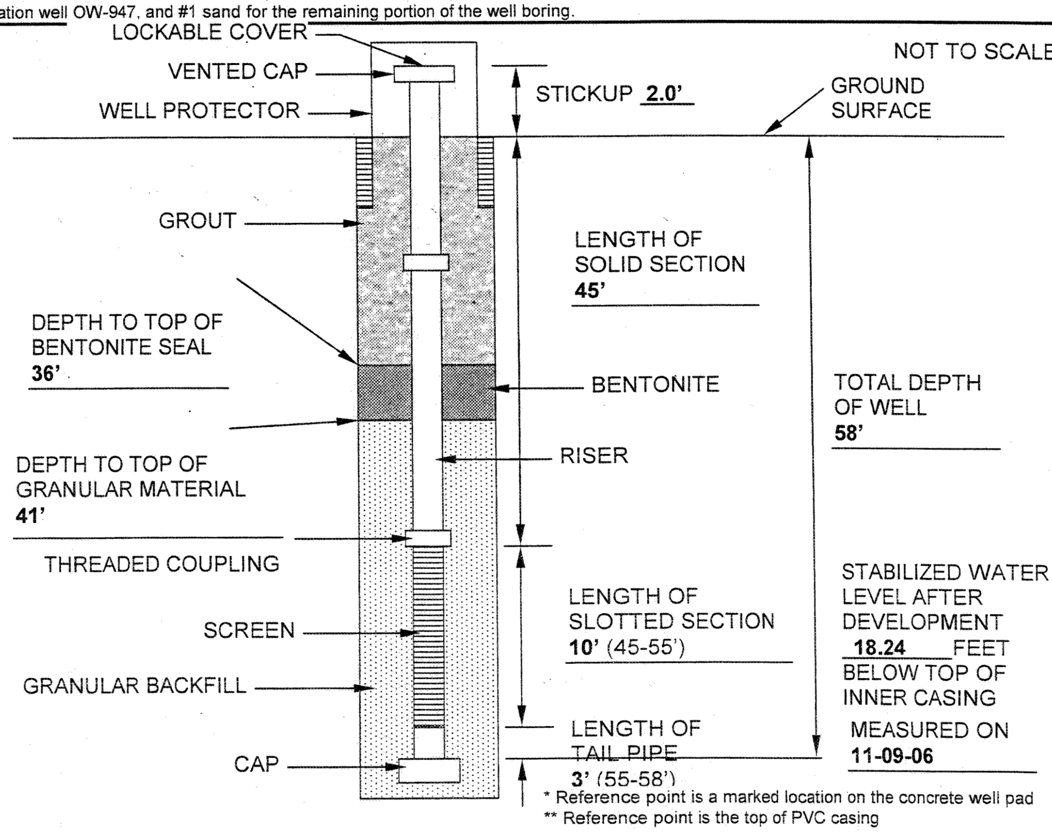


NAPS COL 2.0-30-A **Figure 2.5-276 Log of Well OW-947**

OBSERVATION WELL INSTALLATION RECORD

JOB NAME	NORTH ANNA COL	JOB NUMBER	6468-06-1472
WELL NUMBER	OW-947	INSTALLATION DATE	11-06-06
LOCATION (NAD83)	N = 3,909,579.58 E = 11,686,371.84		
GROUND SURFACE ELEVATION* (NAVD88)	313.30	REFERENCE POINT ELEVATION** (NAVD88)	315.08
GRANULAR BACKFILL MATERIAL	Southern Silica #1 & #3 Sand*	SLOT SIZE	.010
SCREEN MATERIAL	PVC Schd. 40-Standard	SCREEN DIAMETER	2 in.
RISER MATERIAL	PVC Schd. 40-Standard	RISER DIAMETER	2 in.
DRILLING TECHNIQUE	Hollow-stem auger 4.25" I.D.	DRILLING CONTRACTOR	MACTEC
BOREHOLE DIAMETER	Approximately 8"	MACTEC FIELD REPRESENTATIVE	Kim Charles-Smith
LOCK BRAND	Master	SIZE/MODEL	N/A
KEY CODE/COMBINATION	#3206		

* Both #1 and #3 sand met the technical specifications for use as granular backfill material. MACTEC used #3 sand to backfill the sump portion of observation well OW-947, and #1 sand for the remaining portion of the well boring.



* Reference point is a marked location on the concrete well pad
** Reference point is the top of PVC casing

NORTH ANNA POWER STATION MINERAL, VIRGINIA COL PROJECT Dominion Purchase Order 7015798	MACTEC MACTEC Engineering and Consulting, Inc. 3301 Atlantic Avenue Raleigh, North Carolina 27604	OBSERVATION WELL INSTALLATION RECORD [illegible] 1-8-07

Appendix 2A ARCON96 Source/Receptor Inputs

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

2A.2.1 Meteorological Data

Add the following as the last sentence of this section.

NAPS COL 2A.2-1-A

Instrumentation heights used in the analysis are described in SSAR Section 2.3.3.1. Meteorological data from 1996 to 1998 as described in SSAR Section 2.3 is used in the analysis.

2A.2.3 ARCON96 ESBWR Inputs

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

NAPS COL 2A.2-1-A

These directions are adjusted by the difference in angle (approximately 24 degrees counterclockwise) between the ESBWR plant north and the Unit 3 plant north; Unit 3 receptor to source directions are shown in Table 2A-4R.

2A.2.4 Confirmation of the ESBWR χ/Q Values

Replace this section with the following.

NAPS COL 2A.2-1-A

DCD Figure 2A-1 shows the locations of the sources and receptors for ESBWR control room determinations, also used in the Unit 3 evaluations. The dimensions of the diffuse source planes provided in DCD Table 2A-3 are determined as directed by RG 1.194, Regulatory Position 3.2.4.5, for the nearest receptor locations. ARCON96 calculations are performed for source/receptor pairs listed in DCD Table 2A-3 and Table 2A-4R using site-specific meteorological data. Results of the site-specific analysis are provided in Tables 2.3-202 through 2.3-207.

2A.2.5 Confirmation of the Reactor Building χ/Q Values

Replace this section with the following.

NAPS COL 2A.2-2-A

During refueling, doors or personnel air locks on the east sides of the Reactor Building or Fuel Building could act as a point source that could result in control room χ/Q values that are higher than the ESBWR χ/Q values for a release in the Reactor Building. Therefore, the doors are administratively controlled prior to and during movement of irradiated fuel bundles. The administrative controls are such that the doors and personnel air locks on the East sides of the Reactor Building or Fuel Building are promptly closed under conditions indicative of a fuel handling accident.

2A.3 COL Information

2A.2-1-A **Confirmation of the ESBWR χ/Q Values**

NAPS COL 2A.2-1-A This COL item is addressed in Section 2.3.4.3 and in Section 2A.2.4.

2A.2-2-A **Confirmation of the Reactor Building χ/Q Values**

NAPS COL 2A.2-2-A This COL item is addressed in Section 2A.2.5.

NAPS ESP COL 2.3-2
NAPS COL 2A.2-1-A

Table 2A-4R ARCON96 Input – Receptor to Source Direction

Source/Receptor	Receptor to Source Direction (deg.)
RB to CBL	294
RB to EN	284
RB to ES	304
RB to N	308
RB to TSCE	236
RB to TSCW	224
PCCS to CBL	333
PCCS to EN	309
PCCS to ES	328
PCCS to N	332
PCCS to TSCE	238
PCCS to TSCW	225
TB to CBL	7
TB to EN	348
TB to ES	355
TB to N	0
TB to TSCE	256
TB to TSCW	238
TB-TD to CBL	5
TB-TD to EN	355
TB-TD to TSCW	301
FB to CBL	252
FB to EN	258
FB to ES	272
FB to N	276
RW to N	328
RB-VS to CBL	271
RB-VS to ES	285
RB-VS to N	286
TB-VS to CBL	20
TB-VS to EN	5

NAPS ESP COL 2.3-2
NAPS COL 2A.2-1-A

Table 2A-4R ARCON96 Input – Receptor to Source Direction

Source/Receptor	Receptor to Source Direction (deg.)
TB-VS to N	12
RW-VS to CBL	326
RW-VS to EN	314
RW-VS to N	328
BPN to CBL	346
BPN to EN	309
BPN to ES	330
BPN to N	339
BPS to CBL	243
BPS to EN	253
BPS to ES	279
BPS to N	283

Chapter 3 Design of Structures, Components, Equipment, and Systems

3.1 Conformance with NRC General Design Criteria

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

3.2 Classification of Structures, Systems and Components

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

Table 3.2-1 Classification Summary

Replace the note for System P73 with the following.

STD CDI The site-specific plant design includes the HWCS. See Section 9.3.9 for further details.

Replace the note for System P74 with the following.

STD CDI The site-specific plant design does not include the Zinc Injection System.

Replace the note for System U78 with the following.

NA3 CDI The site-specific plant design does not include the cold machine shop.

3.3 Wind and Tornado Loadings

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

3.4 Water Level (Flood) Design

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

3.5 Missile Protection

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

3.5.1.5 Site Proximity Missiles (Except Aircraft)

Add the following sentence after the first sentence in the first paragraph.

STD SUP 3.5-1 Site-specific missile sources are addressed in Section 2.2.

3.5.1.6 Aircraft Hazards

Add the following at the end of the first paragraph.

STD SUP 3.5-2 Site-specific aircraft hazard analysis and the site-specific critical areas are addressed in Section 2.2.

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

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

3.7 Seismic Design

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

3.7.1.1 Design Ground Motion

NAPS SUP 3.7-1

3.7.1.1.4 Site-Specific Design Ground Motion Response Spectra

The site-specific design Ground Motion Response Spectra (GMRS) and the FIRS are described in Section 2.5.2. The CSDRS are compared with the FIRS in Table 2.0-201.

NAPS SUP 3.7-2

3.7.1.1.5 Site-Specific Design Ground Motion Time History

The site-specific earthquake ground motion time history is described in Section 2.5.4.

3.7.1.3 Supporting Media for Seismic Category I Structures

Add the following at the end of the first paragraph.

NAPS SUP 3.7-3 Section 2.5.4 provides site-specific properties of subsurface materials.

3.7.2.4 Soil-Structure Interaction

Add the following at the end of the first paragraph.

NAPS SUP 3.7-4 Section 2.5.4 describes the site-specific properties of subsurface materials.

3.7.2.8 Interaction of Non-Category I Structures with Seismic Category I Structures

Add the following at the end of this section.

NAPS SUP 3.7-5

The locations of structures are provided in Figure 2.1-201.

3.7.4 Seismic Instrumentation

Add the following at the end of the first paragraph.

NAPS SUP 3.7-6

The seismic monitoring program described in this subsection, including the necessary test and operating procedures, will be implemented prior to receipt of fuel on site.

3.8 Seismic Category I Structures

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

3.9 Mechanical Systems and Components

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

3.9.2.4 Initial Startup Flow-Induced Vibration Testing of Reactor Internals

Replace the last two paragraphs with the following.

NAPS COL 3.9.9-1-H

A vibration assessment program as specified in RG 1.20 is provided in DCD Appendix 3L and the following referenced GEH Reports.

- NEDE-33259P, "ESBWR Reactor Internals Flow Induced Vibration Program"
- NEDE-33312P, "Steam Dryer Acoustic Load Definition"
- NEDE-33313P, "Steam Dryer Structural Evaluation"
- NEDC-33408P, "ESBWR Steam Dryer Plant Based Load Evaluation Methodology"

Information on a schedule in accordance with the five applicable scheduling portions of position C.3 of RG 1.20 (refer to Section C.2.5) for non-prototype internals is as follows.

- In response to C.2.5, Item (1), the reactor internals design has been classified by GEH in DCD Section 3L.1 as non-prototype Category II.

- In response to C.2.5, Items (2), (3) and (4), Unit 3 is committed to the comprehensive vibration assessment program including the scope, the vibration measurement and inspection phases and the summary as described in DCD Appendix 3L with no departures.
- In response to C.2.5, Item (5), Unit 3 will submit the preliminary and final reports which together summarize the results of the vibration analysis, measurement, and inspection programs to the NRC within 60 days and 180 days, respectively, following the completion of the vibration testing.

3.9.3.1 Loading Combinations, Design Transients and Stress Limits

Replace the last sentence with the following.

STD COL 3.9.9-2-H

The piping stress reports identified in this DCD section will be completed within six months of completion of ITAAC Table 3.1-1. The FSAR will be revised as necessary in a subsequent update to address the results of this analysis.

3.9.3.7.1(3)e Snubber Preservice and Inservice Examination and Testing

Preservice Examination and Testing

Add the following at the end of this section.

STD COL 3.9.9-4-A

A preservice thermal movement examination is also performed; during initial system heatup and cooldown, for systems whose design operating temperature exceeds 121°C (250°F), snubber thermal movement is verified.

Additionally, preservice operational readiness testing is performed on all 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.

The preservice examination and testing programs for snubbers will be completed in accordance with milestones described in Section 13.4.

Inservice Examination and Testing

Add the following at the beginning of this section.

STD COL 3.9.9-4-A

Inservice examination and testing of all 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 will be 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 then current interval.

An inservice visual examination is performed on all 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 scheduled start of the applicable 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 are 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 an 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 ensure that test parameters that may have been affected by the repair or maintenance activity are verified acceptable.

Service life for snubbers is established, monitored and adjusted as required by ISTD-6000 and the guidance of ASME OM Code Nonmandatory Appendix F.

The inservice inspection and testing programs for snubbers will be completed in accordance with milestones described in Section 13.4.

Delete the last two sentences of the last paragraph.

3.9.3.7.1(3)f Snubber Support Data

Replace the first sentence with the following.

STD COL 3.9.9-4-A

For the ASME Class 1, 2, and 3 systems listed in DCD Tier 1, Section 3.1, that contain snubbers, a plant-specific table will be prepared in conjunction with the closure of the system-specific ITAAC for piping and component design and will include the following specific snubber information.

Add the following at the end of this section.

STD COL 3.9.9-4-A

This information will be included in the FSAR as part of a subsequent FSAR update.

3.9.6 Inservice Testing of Pumps and Valves

Replace the last sentence of the last paragraph with the following.

STD COL 3.9.9-3-A

Milestones for implementation of the ASME OM Code preservice and inservice testing programs are defined in Section 13.4.

3.9.6.1 Inservice Testing of Valves

Add the following before the last paragraph.

STD COL 3.9.9-3-A

Each valve subject to inservice testing is also tested during the preservice test (PST) 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, or valves that are 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.

3.9.6.1.4 Valve Testing

Add the following at the end of the introduction to this section.

STD COL 3.9.9-3-A

Other specific testing requirements for power-operated valves include stroke-time testing and, as applicable, diagnostic testing to evaluate valve condition and to verify the valve will continue to function under design-basis conditions.

(1) Valve Exercise Tests

Add the following after the second sentence of the first paragraph.

STD COL 3.9.9-3-A

Valves are tested by full-stroke exercising, during operation at power, to the positions required to fulfill their functions.

Add the following after the third sentence of the first paragraph.

STD COL 3.9.9-3-A

If full-stroke exercising is not practicable, part-stroke exercising is performed during operation at power or during cold shutdown.

Add the following new paragraph after the first paragraph.

STD COL 3.9.9-3-A During extended shutdowns, valves that are required to be operable must remain capable of performing their intended safety function. Exercising valves during cold shutdown commences within 48 hours of achieving cold shutdown and continues until testing is complete or the plant is ready to return to operation at power. Valve testing required to be performed during a refueling outage is completed before returning the plant to operation at power.

Add the following after the first sentence of the second paragraph.

STD COL 3.9.9-3-A Valve testing uses reference values determined from the results of PST or IST. These tests that establish reference values are performed under conditions as near as practicable to those expected during the IST. Stroke time is measured and compared to the reference value, except for valves classified as fast-acting (e.g., solenoid-operated valves (SOVs) with stroke time less than 2 seconds), for which a stroke time limit of 2 seconds is assigned.

Add the following after the third paragraph.

STD COL 3.9.9-3-A SOVs are tested to confirm the valves move to their energized positions and are maintained in those positions, and to confirm that the valves move to the appropriate failure mode positions when de-energized.

Pre-conditioning of valves or their associated actuators or controls prior to IST undermines the purpose of IST and is prohibited. 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.

3.9.6.1.5 Specific Valve Test Requirements

(1) Power-Operated Valve Tests

Replace the last paragraph with the following.

STD COL 3.9.9-3-A Section 3.9.6.8 describes additional (non-Code) testing of power-operated valves as discussed in Regulatory Issue Summary 2000-03.

(3) Check Valve Exercise Tests

Add the following as the first sentence of the second paragraph.

STD COL 3.9.9-3-A

Check valve testing requires verification that obturator movement is in the direction required for the valve to perform its safety function.

Add the following before the last paragraph.

STD COL 3.9.9-3-A

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.

If these test methods are impractical for certain check valves, or if sufficient flow cannot be achieved or verified, a sample disassembly examination program verifies valve obturator movement. The sample disassembly examination program groups check valves by category of similar design, application, and service condition.

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.

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 practical, 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.

Preoperational testing is performed during the initial test program (refer to Section 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.

Data acquired during check valve testing and inspections, and the maintenance history of a valve or group of valves is collected and maintained in order to establish the basis for specifying inservice testing, examination, and preventive maintenance activities that will identify and/or mitigate the failure of the check valves or groups of check valves tested. This data is also used to determine if certain check valve condition monitoring tests, such as nonintrusive tests, are feasible and effective in monitoring for these identified failure mechanisms, whether periodic disassembly and examination activities would be effective in monitoring for these failure mechanisms, as well as to determine possible valve groupings to implement in a future check valve condition monitoring program as allowed by ISTC-5222, the requirements of which are described in ASME OM Code, Appendix II.

3.9.6.5 Valve Replacement, Repair and Maintenance

Add the following to the end of the paragraph.

STD COL 3.9.9-3-A

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.

3.9.6.6 10 CFR 50.55a Relief Requests and Code Cases

Add the following at the end of the first paragraph.

STD SUP 3.9-1

No relief from or alternative to the ASME OM Code is being requested.

3.9.6.7 Inservice Testing Program Implementation

Delete the last paragraph.

3.9.6.8 Non-Code Testing of Power-Operated Valves

Replace the second sentence of the first paragraph with the following.

STD COL 3.9.9-3-A

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.

Replace the fifth sentence of the first paragraph with the following.

STD COL 3.9.9-3-A

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.

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

STD COL 3.9.9-3-A

Uncertainties affecting both valve function and structural limits are addressed.

Replace the second paragraph with the following.

STD COL 3.9.9-3-A

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 3.9.201 and 3.9.202). The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in RIS 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-related Power-Operated Valves Under Design Basis Conditions, (Reference 3.9.203) 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 3.9.201 and 3.9.202, 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, valve repair or replacement, have the potential to affect valve functional performance.
- Guidance is included to address lessons learned from other valve programs in procedures and training 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.

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.

3.9.7 Risk-Informed Inservice Testing

Replace this section with the following.

STD SUP 3.9-2 Risk informed inservice testing is not being utilized.

3.9.8 Risk-Informed Inservice Inspection of Piping

Replace this section with the following.

STD SUP 3.9-3 Risk informed inservice inspection is not being utilized.

3.9.9 COL Information

3.9.9-1-H Reactor Internals Vibration Analysis, Measurement and Inspection Program

NAPS COL 3.9.9-1-H This COL item is addressed in Section 3.9.2.4.

3.9.9-2-H ASME Class 2 or 3 or Quality Group D Components with 60 Year Design Life

STD COL 3.9.9-2-H This COL item is addressed in Section 3.9.3.1.

3.9.9-3-A Inservice Testing Programs

STD COL 3.9.9-3-A This COL item is addressed in Section 3.9.6.

3.9.9-4-A Snubber Inspection and Test Program

STD COL 3.9.9-4-A This COL item is addressed in Section 3.9.3.7.1(3)e and Section 3.9.3.7.1(3)f.

3.9.10 References

3.9.201 Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000.

3.9.202 USNRC, Eugene V. Imbro, letter to Mr. David J. Modeen, Nuclear Energy Institute, Comments On Joint Owners' Group Air Operated Valve Program Document, October 8, 1999.

3.9.203 Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-related Power-Operated Valves Under Design Basis Conditions, March 15, 2000.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

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

3.10.1.4 Dynamic Qualification Report

Replace the last paragraph with the following.

STD COL 3.10.4-1-A A schedule will be provided within 12 months after issuance of the COL that supports planning for and conducting of NRC inspections of seismic and dynamic qualification of mechanical and electrical equipment. The schedule will be updated every 6 months until 12 months before scheduled fuel loading.

The Dynamic Qualification Report will be completed prior to fuel load. FSAR information will be revised, as necessary, as part of a subsequent FSAR update.

STD SUP 3.10-1 Section 17.5 defines the Quality Assurance Program requirements that are applied to equipment qualification files, including requirements for handling safety-related quality records, control of purchased material, equipment and services, test control, and other quality related processes.

3.10.4 COL Information

3.10.4-1-A Dynamic Qualification Report

STD COL 3.10.4-1-A This COL item is addressed in Section 3.10.1.4.

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.

3.11.4.4 Environmental Qualification Documentation

Replace the last paragraph with the following.

STD COL 3.11-1-A A description of the environmental qualification program is provided in DCD Section 3.11.

Implementation of the environmental qualification program, including development of the plant specific Environmental Qualification Document (EQD), will be in accordance with the milestone defined in Section 13.4.

3.11.7 COL Information

3.11-1-A Environmental Qualification Document

STD COL 3.11-1-A This COL item is addressed in Section 3.11.4.4.

STD SUP 3.12-1

3.12 Piping Design Review

Information on seismic Category I and II, and nonseismic piping analysis and their associated supports is presented in DCD Sections 3.7, 3.9, 3D, 3K, 5.2 and 5.4.

STD SUP 3.12-2 The location and distance between piping systems will be established as part of the completion of ITAAC Table 3.1-1. The FSAR will be revised as necessary, in a subsequent update to include this information.

STD SUP 3.13-1

3.13 Threaded Fasteners - ASME Code Class 1, 2, and 3

Criteria applied to the selection of materials, design, inspection and testing of threaded fasteners (i.e., threaded bolts, studs, etc.) are presented in DCD Section 3.9.3.9, with supporting information in DCD Sections 4.5.1, 5.2.3, and 6.1.1.

Appendix 3A Seismic Soil-Structure Interaction Analysis

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

3A.1 Introduction

Replace the last sentence in the second paragraph with the following.

NAPS CDI Site-specific geotechnical data is described in Chapter 2. This data is compatible with the site enveloping parameters considered in the standard design.

3A.2 ESBWR Standard Plant Site Plan

Replace the first two sentences of the first paragraph with the following.

NAPS CDI The site plan is shown in Figure 2.1-201. The plan orientation is denoted on the figure.

Appendix 3B Containment Hydrodynamic Load Definitions

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

Appendix 3C Computer Programs Used in the Design and Analysis of Seismic Category I Structures

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

Appendix 3D Computer Programs Used in the Design of Components, Equipment, and Structures

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

Appendix 3E [Deleted]

Appendix 3F Response of Structures to Containment Loads

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

Appendix 3G Design Details and Evaluation Results of Seismic Category I Structures

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

Appendix 3H Equipment Qualification Design Environmental Conditions

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

Appendix 3I Designated NEDE-24326-1-P Material Which May Not Change Without Prior NRC Approval

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

Appendix 3J Evaluation of Postulated Ruptures in High Energy Pipes

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

Appendix 3K Resolution of Intersystem Loss of Coolant Accident

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

Appendix 3L Reactor Internals Flow Induced Vibration Program

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

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 the following departures and/or supplements.

4.3.3.1 Nuclear Design Description

Replace the last paragraph with the following.

STD COL 4.3-1-A There are no changes to the fuel, control rod, or core design from that described in the referenced certified design.

4.3.5 COL Information

4.3-1-A Variances from Certified Design

STD COL 4.3-1-A This COL Item is addressed in Section 4.3.3.1.

4.4 Thermal and Hydraulic Design

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

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 System

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

Appendix 4A Typical Control Rod Patterns and Associated Power Distribution for ESBWR

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

4A.1 Introduction

Replace the third paragraph with the following.

STD COL 4A-1-A There are no changes to the fuel, control rod, or core design from that described in the referenced certified design.

4A.3 COL Information

4A-1-A Variances from Certified Design

STD COL 4A-1-A This COL item is addressed in Section 4A.1.

Appendix 4B Fuel Licensing Acceptance Criteria

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

Appendix 4C Control Rod Licensing Acceptance Criteria

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

Appendix 4D Stability Evaluation

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

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 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50.55a

Add the following at the end of this section.

STD SUP 5.2-2

As described in Section 5.2.4, preservice and 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, required by 10 CFR 50.55a. As described in DCD Section 3.9.6 for pumps and valves, and in DCD Section 3.9.3.7.1 for dynamic restraints, preservice and inservice testing of the reactor coolant pressure boundary components is in accordance with the edition and addenda of the ASME OM Code required by 10 CFR 50.55a.

5.2.1.2 Applicable Code Cases

Add the following as the third sub-bulleted paragraph after the second sub-bullet of the third bullet in the first paragraph.

STD SUP 5.2-3

— Regulatory Guide 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code.” This guide lists those ASME OM Code cases that are acceptable to the staff for use in the preservice and inservice testing of pumps, valves, and dynamic restraints in light-water-cooled nuclear power plants.

5.2.4 Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

Replace the second sentence in the second paragraph with the following.

STD COL 5.2-3-A

All Class 1 austenitic or dissimilar metal welds are included in the referenced certified design.

Replace the second sentence and subsequent parenthetical sentence in the fourth paragraph with the following.

STD COL 5.2-1-A

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.

5.2.4.2 Accessibility

Replace the last sentence in the second paragraph with the following.

STD COL 5.2-3-A

During the construction phase of the project, anomalies and construction issues are addressed using change control procedures. Procedures require that changes to approved design documents, including field changes and modifications, are subject to the same review and approval process as the original design. Accessibility and inspectability are key components of the design process. Control of accessibility for inspectability and testing during licensee design activities affecting Class I components is provided via procedures for design control and plant modifications.

Ultrasonic techniques (UT) will be the preferred NDE method for all PSI and ISI volumetric examinations; radiographic techniques (RT) will be used as a last resort only if UT cannot achieve the necessary coverage. The same NDE method used during PSI will be used for ISI to the extent possible to assure a baseline point of reference. If a different NDE method is used for ISI than was used for PSI, equivalent coverage will be achieved as required by code.

5.2.4.3.4 Qualification of Personnel and Examination Systems for Ultrasonic Examination

Add the following at the end of the paragraph.

STD COL 5.2-1-A

Certification of NDE personnel shall be in accordance with ASME Section XI, IWA-2300, as modified by 10 CFR 50.55a(b)(2)(xviii).

5.2.4.6 System Leakage and Hydrostatic Pressure Tests

Revise the second sentence of the first paragraph as follows.

STD COL 5.2-1-A

Regardless of which test method is chosen, system leakage and hydrostatic pressure tests will meet all requirements of ASME Code

Section XI, IWA-5000 and IWB-5000 for Class I components, including the limitation of 10 CFR 50.55a(b)(2)(xxvi).

Add the following paragraph at the end of this section.

STD SUP 5.2-1

System pressure tests and correlated technical specification requirements are provided in the plant Technical Specifications 3.4.4, "RCS Pressure and Temperature (P/T) Limits," and 3.10.1, "Inservice Leak and Hydrostatic Testing Operation."

5.2.4.11 COL Information for Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

Replace the first sentence of the first paragraph with the following and delete the last sentence.

STD COL 5.2-1-A

DCD Section 5.2.4 fully describes the Preservice and Inservice Inspection and Testing Programs for the RCPB. The implementation milestones for the Preservice and Inservice Inspection and Testing Programs are provided in Section 13.4.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

STD COL 5.2-2-H

Delete the parenthetical statement in the first sentence of the first paragraph.

Replace DCD Section 5.2.5.9 with the following.

STD COL 5.2-2-H

5.2.5.9 Leak Detection Monitoring

Operators are provided with procedures for detecting, monitoring, recording, trending, and determining the sources of reactor coolant pressure boundary leakage. Examples of parameters that are monitored are sump pump run time, sump level, condensate transfer rate, and process chemistry/radioactivity.

The procedures are used for converting different parameter indications for identified and unidentified leakage into common leak rate equivalents (volumetric or mass flow) and leak rate rate-of-change values, including indications from: 1) the drywell floor drain high conductivity water sump monitoring system, 2) the drywell air coolers condensate flow monitoring system, and 3) the drywell fission product monitoring system.

The procedures are used to monitor leakage at levels well below Technical Specifications limits and provide guidance for evaluating potential corrective action plans to prevent the plant from exceeding a Technical Specifications limit.

An unidentified leakage rate-of-change alarm provides an early alert to the operators to initiate corrective actions prior to reaching a Technical Specifications limit.

A description of the plant procedures program and implementation milestones are provided in Section 13.5.

5.2.6 COL Information

5.2-1-A Preservice and Inservice Inspection Program Description

STD COL 5.2-1-A This COL Item is addressed in Sections 5.2.4, 5.2.4.3.4, 5.2.4.6, 5.2.4.11, and 6.6.

5.2-2-H Leak Detection Monitoring

STD COL 5.2-2-H This COL Item is addressed in Sections 5.2.5 and 5.2.5.9.

5.2-3-A Preservice and Inservice Inspection NDE Accessibility Plan Description

STD COL 5.2-3-A This COL Item is addressed in Section 5.2.4 and 5.2.4.2.

5.3 Reactor Vessel

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

5.3.1.5 Fracture Toughness

Compliance with 10 CFR 50, Appendix G

Replace the last sentence in the first paragraph with the following.

STD COL 16.0-2-H 5.6.4-1 The pressure-temperature limit curves are developed in accordance with the Pressure and Temperature Limits Report, as discussed in the Technical Specifications Section 5.6.4.

5.3.1.8 COL Information for Reactor Vessel Material Surveillance Program

Replace this section with the following.

STD COL 5.3-2-A

The description of the reactor vessel material surveillance program in DCD Section 5.3.1.6 is supplemented as follows.

A complete reactor vessel material surveillance program will be developed as described above in accordance with the implementation schedule provided in Section 13.4.

5.3.1.8.1 Locations of Capsules in Core Beltline Region

A total of four irradiation exposure specimen sets containing the required specimens are located near the vessel wall slightly above the core midplane. The irradiation exposure specimen sets are contained in specimen holders that are welded to the inner diameter of the core beltline forging. Each specimen holder houses two specimen containers that form the irradiation exposure set. The elevation and azimuth locations of the exposure specimen sets align with the maximum calculated fluence within the core beltline. Based on the location of the samples relative to the shell forging and their placement at the peak fluence location, the lead factors for the samples will be greater than 1.0. The lead factor for the specimens when placed at the peak location has been estimated to be 1.17.

5.3.1.8.2 Preparation of Capsule Specimens

As stated in DCD Section 5.3.1.6.1, the reactor vessel materials specimens are provided in accordance with the requirements of ASTM E 185 and 10 CFR 50, Appendix H. The surveillance specimen materials are prepared from full thickness samples taken from the actual core beltline forging and from the adjacent forgings and weld materials. The materials include the base metal and weld metal that have the highest adjusted reference temperature at end-of-life. The fabrication or heat treatment history (austenitizing, quench and tempering, and post-weld heat treatment) of the test material is fully representative of the fabrication history of the materials in the beltline of the RPV.

The base metal sample blocks from which the specimens are taken are located at least one “T” from any quenched edge of the block, where “T” is the material thickness, and at least 25 mm from a flame cut edge or weld fusion line.

The weld metal sample blocks are fabricated using the same welding procedure and process as the vessel shell weld they represent. The welding materials (electrodes, flux, or gas) are from the same heat and lot as the material used to make the production weld. The welder is qualified to ASME Section IX. The weld must satisfy the same examination and inspection requirements as the production weld. The weld or HAZ samples are taken at least one "T" from any quenched edge of the block, at least 25 mm from a flame cut edge, and at least 13 mm from the root of the weld.

Base Metal Samples

The longitudinal axes of tensile specimens are located 1/4T from the as-quenched vessel surface. The specimens are oriented so that the longitudinal axis is parallel to the forging and normal to the major working direction of the forging.

Charpy V-notch specimens are removed 1/4T from the as-quenched vessel surface. The longitudinal axes of specimens are oriented parallel to the forging surface and normal to the major working direction.

Weld Metal Samples

The longitudinal axes of tensile specimens are located in the approximate center of the weld metal and at least 13 mm from the final weld surface and the root of the weld. The axis is parallel to the plate or forging surface.

The roots of the notch of Charpy V-notch specimens are in the approximate center of the weld metal. The specimens are taken at least 13 mm from the final weld surface and the root of the weld. The notch is perpendicular to the plate or forging surface.

All tensile specimens and Charpy V-notch specimens correspond to the allowable specimen types, as defined in ASTM E 185.

Fracture Toughness Samples

Fracture toughness specimens are provided from the limiting base and weld metals and are consistent with the guidelines in ASTM E 1820 and ASTM E 1921.

5.3.1.8.3 Number and Type of Specimens

The number of specimens in each exposure set satisfies or exceeds the requirements of ASTM E 185. Additional fracture toughness specimens of the limiting materials are included as shown in Table 5.3-201. Four sets of specimens are provided for the 60-year life of the ESBWR. The quantities of specimens per irradiation exposure set are provided in Table 5.3-201.

5.3.1.8.4 Report of Test Results

A summary technical report, including test results, is submitted as specified in 10 CFR 50.4, for the contents of each capsule withdrawn, 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 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, the expected date for submittal of the revised Technical Specification will be provided with the report.

5.3.3.6 Operating Conditions

Add the following after the first sentence.

STD SUP 5.3-1

Development of plant operating procedures is addressed in Section 13.5. These procedures require compliance with the Technical Specifications. The Technical Specifications (which are developed by the methodology also identified in the Technical Specifications) are intended to ensure that the P-T limits identified in DCD Section 5.3.2 are not exceeded during normal operating conditions and anticipated plant transients.

5.3.4 COL Information

5.3-2-A Materials and Surveillance Capsule

STD COL 5.3-2-A

This COL Item is addressed in Section 5.3.1.8.

5.4 Component and Subsystem Design

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

5.4.8 Reactor Water Cleanup/Shutdown Cooling System

Add the following paragraph at the end of this section.

STD SUP 5.4-1

Operating procedures provide guidance to prevent severe water hammer caused by mechanisms such as voided lines.

5.4.12 Reactor Coolant System High Point Vents

Add the following paragraph at the end of this section.

STD SUP 5.4-2

A human factors analysis of the control room displays and controls for the RCS vents is included as part of the overall human factors analysis of the control room displays and controls described in DCD Chapter 18. This analysis considers:

- The use of this information by an operator during both normal and abnormal plant conditions;
- Integration into emergency procedures;
- Integration into operator training; and
- Other alarms during an emergency and the need for prioritization of alarms.

5.4.12.1 Operation of RPV Head Vent System

Add the following paragraph at the end of this section.

STD SUP 5.4-3

Operating procedures for the reactor vent system address considerations regarding when venting is needed and when it is not needed, including a variety of initial conditions for which venting may be required. The development of operating procedures is addressed in Section 13.5.

STD COL 5.3-2-A

Table 5.3-201 Quantities of Reactor Vessel Materials Specimens per Irradiation Exposure Set

Material	Specimen Type	No. of Specimens per Irradiation Exposure Set	Comments
Base Metal	Charpy	45	15 samples from each of three forgings in accordance with ASTM E 185-02.
	Tensile	9	3 samples from each of three forgings in accordance with ASTM E 185-02.
	Fracture Toughness	8	Taken from most limiting material in accordance with ASTM E 185-02.
Weld Metal	Charpy	30	15 specimens per weld in accordance with ASTM E 185-02.
	Tensile	6	3 specimens per weld in accordance with ASTM E 185-02.
	Fracture Toughness	8	Taken from most limiting material in accordance with ASTM E 185-02.
HAZ	Charpy	12	In accordance with ASTM E 185-82.

Chapter 6 Engineered Safety Features

6.0 General

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

6.1 Design Basis Accident Engineered Safety Feature Materials

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

6.2 Containment Systems

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

6.2.1.6 Testing and Inspection

Add the following at the end of this section.

STD SUP 6.2-1

Inspections to Limit Debris

Procedures describe the activities necessary to prevent debris from affecting the emergency core cooling and long-term cooling safety functions in accordance with RG 1.82, including: 1) inspection of the cleanliness of pools within containment, 2) a visual examination for evidence of structural degradation or corrosion of debris screens, 3) an inspection of the wetwell and the drywell, including the vents, downcomers, and deflectors, for the identification and removal of debris or trash that could contribute to the blockage of debris screens for the ECC and long-term cooling safety functions, 4) containment cleanliness programs to clean the pools within containment on a regular basis, and 5) plant procedures for control and removal of foreign materials from the containment and abatement procedures to avoid latent debris generation during removal and/or replacement of insulation within containment.

6.2.4.2 System Design

Replace the fourth sentence in the first paragraph with the following.

STD COL 6.2-1-H

DCD Tables 6.2-16 through 6.2-45 require an entry for the length of pipe from the containment to the inboard and outboard isolation valves. Pipe lengths will be determined as part of completion of the piping design ITAAC identified in DCD Tier 1, Table 3.1-1. The FSAR will be revised to reflect the pipe length information in a subsequent update.

6.2.8 COL Information

6.2-1-H Pipe Length from Containment to Inboard/Outboard Isolation Valve

STD COL 6.2-1-H This COL item is addressed in Section 6.2.4.2.

6.3 Emergency Core Cooling Systems

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

6.4 Control Room Habitability Systems

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

6.4.4 System Operation Procedures

Replace the second paragraph with the following.

STD COL 6.4-1-A Operators are provided with training and procedures for control room habitability that address the applicable aspects of NRC Generic Letter 2003-01 and are consistent with the intent of Generic Issue 83. Training and procedures are developed and implemented in accordance with Sections 13.2 and 13.5, respectively. The implementation milestones for training and procedures are provided in Sections 13.4 and 13.5, respectively.

6.4.5 Design Evaluations

System Safety Evaluation

Add the following after the second paragraph.

NAPS SUP 6.4-1 The impact of a postulated design basis accident (DBA) in Units 1 or 2 on the Unit 3 control room was evaluated. The bounding case is a release from the Unit 2 RB to the Unit 3 Control Building receptor based on a minimum distance criterion. The evaluation was performed as follows:

- Atmospheric dispersion factors, χ/Qs, at the Unit 3 MCR intakes were conservatively calculated assuming a point source, a distance of approximately 400 m (1312 ft), and a release height of 10 m (32.8 ft).

Meteorological data used for cross-unit impact is consistent with that used for the χ/Q values presented in Section 2.3. A nominal "receptor to source" direction of 60 degrees was assumed (clockwise with respect to "true north"). The χ/Q values are presented in Table 2.3-207.

- The Unit 2 LOCA as described in Section 15.4.1.8 of the Units 1 and 2 UFSAR was reviewed. The resultant dose at the Unit 3 MCR intake was determined by adjusting the LPZ dose consequences by the ratio of the χ/Q values, and the ratio of the breathing rates (BR) for the LPZ versus the control room values. Detailed modeling of the Unit 3 control room was not performed because the doses are bounded by a postulated Unit 3 LOCA. No credit was taken for the reduced control room occupancy factor, the Unit 3 control room emergency filtration units, or the "finite cloud" model allowed per RG 1.194.

Based on this conservative analysis, the resultant dose is bounded by the control room operator dose from a postulated Unit 3 DBA, and is less than GDC 19 limits.

Replace DCD Table 6.4-2 with Table 2.2-202, replace the third paragraph with the following, and delete the last paragraph.

NAPS COL 6.4-2-A

Potential toxic gas sources are evaluated to confirm that an external release of hazardous chemicals does not impact control room habitability. These sources include: 1) offsite industrial facilities and transportation routes; 2) Units 1 and 2; and 3) Unit 3.

Evaluation of potentially hazardous off-site chemicals within 8 km (5 miles) of the control room is addressed in Section 2.2. As described therein, there are no manufacturing plants, chemical plants, storage facilities, major water transportation routes, oil pipelines or gas pipelines within 8 km (5 miles) of the control room. There are also no significant control room habitability impacts due to chemicals being transported along offsite routes within 8 km (5 miles) of the plant.

Toxic gas analysis for potentially hazardous chemicals stored on site is performed in accordance with the guidelines of RG 1.78 and on the basis of no action being taken by the control room operator. The results of the analysis, when compared to the toxicity limits given in RG 1.78 and National Air Quality Standards, show hazardous concentrations of toxic gas in the control room are not reached.

On-site locations with potentially toxic chemicals are identified in Table 2.2-202.

Hydrogen and oxygen storage facilities are in excess of 230 meters (750 ft) from the control room. This distance is acceptable for toxic gas concerns per RG 1.78 based on hazards of postulated instantaneous release followed by vapor cloud explosion or intake of a flammable vapor concentration into a safety-related intake. The hazard for the oxygen supply was a postulated release with an increased concentration at a safety related intake. Calculations performed to evaluate the habitability of the control room for accidental releases of hydrogen or oxygen from the HWCS indicate control room personnel are not subject to the hazard of breathing air with insufficient oxygen inside the control room due to a release of hydrogen. Other identified chemicals are stored in amounts and locations that are adequately separated from the control room intakes such that detection and/or control room isolation is not required.

The maximum concentrations for on-site chemicals, as calculated for Units 1 and 2, are based on the equations provided in NUREG-0570. This evaluation is bounding for the Unit 3 control room intake on the basis of a greater separation distance from Unit 1 and 2 control rooms than the Unit 3 control room. The relative locations for the chemical storage areas, as well as the control room intakes and refresh rates for Unit 1/2 and Unit 3 were considered in the analysis along with the properties of the stored chemicals. The maximum concentrations determined for the room intakes were evaluated for safety in comparison with the toxicity limits from RG 1.78. The analysis performed shows that the control room concentration for a given chemical does not exceed the applicable toxicity limit. Based on this analysis, Seismic Category I Class safety-related toxic gas monitoring instrumentation is not required.

6.4.9 COL Information

6.4-1-A CRHA Procedures and Training

STD COL 6.4-1-A This COL item addressed in Section 6.4.4.

6.4-2-A Toxic Gas Analysis

NAPS COL 6.4-2-A This COL item addressed in Section 6.4.5 and Table 2.2-202.

6.5 Atmosphere Cleanup Systems

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

6.6 Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping

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

STD COL 6.6-2-A Delete the second sentence in the third paragraph.

Replace the last three sentences and the parenthetical statement of the fourth paragraph with the following.

STD COL 6.6-1-A The PSI/ISI program description for Class 2 and 3 components and piping is provided in DCD Section 6.6.

6.6.2 Accessibility

Replace the last sentence in the second paragraph with the following.

STD COL 6.6-2-A All Class 2 or 3 austenitic or dissimilar metal welds are included in the referenced certified design.

During the construction phase of the project, anomalies and construction issues are addressed using change control procedures. Procedures require that changes to approved design documents, including field changes and modifications, are subject to the same review and approval process as the original design.

Accessibility and inspectability are key components of the design process. Control of accessibility for inspectability and testing during licensee design activities affecting Class 2 and 3 components is provided via procedures for design control and plant modifications.

UT will be the preferred NDE method for all PSI and ISI volumetric examinations; RT will be used as a last resort only if UT cannot achieve the necessary coverage. The same NDE method used during PSI will be used for ISI to the extent possible to assure a baseline point of reference. If a different NDE method is used for ISI than was used for PSI, equivalent coverage will be achieved as required by code.

6.6.6 System Pressure Tests

Revise the second sentence of the first paragraph as follows.

STD COL 5.2-1-A

Regardless of which test method is chosen, system leakage and hydrostatic pressure tests will meet all applicable requirements of ASME Code Section XI, IWA-5000 and IWC-5000 for Class 2 components; and IWD-5000 for Class 3 components, including the limitations of 10 CFR 50.55a(b)(2)(xx) and 10 CFR 50.55a(b)(2)(xxvi).

6.6.7 Augmented Inservice Inspections

STD COL 6.6-1-A

6.6.7.1 Flow Accelerated Corrosion Program Description

The flow accelerated corrosion (FAC) monitoring program analyzes, inspects, monitors, and trends nuclear power plant piping and components that are susceptible to FAC damage. The FAC program is based on EPRI NSAC-202L (Reference 6.6-201).

Prior to start-up, a comprehensive FAC-susceptibility screening will be performed to identify any plant systems that may be susceptible to FAC degradation. Should any plant systems remain susceptible, a FAC program will be implemented as described below. Program implementation milestones are provided in Section 13.4. Pre-service baseline nondestructive examination (NDE) inspections will be performed and material constituency identified for each as-fabricated piping component in the susceptible systems.

6.6.7.1.1 Analysis

A program similar to that described in EPRI NSAC-202L 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. For each piping component, the program predicts the wear, and the estimated time until it must be re-inspected, repaired, or replaced.

6.6.7.1.2 Industry Experience

Industry experience provides a valuable supplement to plant analysis and associated inspections. Reviews of industry experience are performed to identify generic plant problem areas and determine differences in similar types of components. This information is used to update the FAC program.

6.6.7.1.3 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 preservice inspections are used as a baseline for later inspections. Therefore, the preservice inspections use grid locations and measurement methods most likely to be used for inservice inspections according to industry guidelines. Each subsequent inspection determines the wear rate for the piping and components and the need for inspection frequency adjustment for those components.

6.6.7.1.4 Training and Engineering Judgement

The FAC program is administered by trained and experienced personnel. Task-specific training is provided for plant personnel that implement the monitoring program. Specific 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.

6.6.7.1.5 Long-Term Strategy

The FAC program includes a long-term strategy that focuses on reducing wear rates, using improved water chemistry, and optimizing the inspection planning process.

6.6.7.1.6 FAC Program Documentation

A procedure documents the overall program description and its implementation.

Governing Program Description

A governing program description defines the overall program and associated responsibilities. This program description addresses the following elements:

- A corporate commitment to monitor and control FAC.
- Identification of the tasks to be performed (including implementing procedures) and associated responsibilities.

- Identification of the position that has overall responsibility for the FAC program.
- Communication requirements between the lead position 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.
- A method for evaluating plant performance against long-term goals.

Program Implementation

The implementation of each specific task conducted as part of the FAC program is described in one or more procedures, including:

- Identifying susceptible systems
- Developing FAC inspection drawings
- Developing a FAC inspection database
- Performing FAC analysis
- Selecting and scheduling components for initial inspection
- Performing inspections
- Evaluating inspection data
- Evaluating worn components
- Identifying components for repair and replacement when necessary
- Selecting and scheduling locations for follow-on inspections

6.6.7.1.7 Documentation

The results of inspections are documented in accordance with the requirements of the implementing documents. Periodically, reports are prepared that identify the components inspected, justify the basis for their selection (i.e., predictive ranking, industry experience, engineering judgment), document the results of the inspections, and evaluate and disposition worn components.

6.6.10 Plant Specific PSI/ISI Program Information

6.6.10.1 Relief Requests

Add the following at the end of this section.

STD COL 6.6-1-A No relief requests for the PSI/ISI program have been identified.

6.6.10.2 Code Edition

Replace the second sentence with the following.

STD COL 6.6-1-A

The initial ISI program incorporates the latest edition and addenda of the ASME Code approved in 10 CFR 50.55a(b) on the date 12 months before initial fuel load.

STD COL 6.6-1-A

6.6.10.3 Program Implementation

The milestones for preservice and inservice inspection program implementation are provided in Section 13.4.

6.6.11 COL Information

6.6-1-A PSI/ISI Program Description

STD COL 6.6-1-A

This COL item is addressed in Section 6.6.

6.6-2-A PSI/ISI NDE Accessibility Plan Description

STD COL 6.6-2-A

This COL item is addressed in Section 6.6.2.

6.6.12 References

6.6-201 Electric Power Research Institute, "Recommendations for an Effective Flow-Accelerated Corrosion Program," NSAC-202L-R2.

Appendix 6A TRACG Application for Containment Analysis

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

Appendix 6B Evaluation of the TRACG Nodalization for the ESBWR Licensing Analysis

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

Appendix 6C Evaluation of the Impact of Containment Back Pressure On the ECCS Performance

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

Appendix 6D Containment Passive Heat Sink Details

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

Appendix 6E TRACG LOCA Containment Response Analysis

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

Appendix 6F Break Spectrums of Break Sizes and Break Elevations

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

Appendix 6G TRACG LOCA SER Confirmation Items

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

Appendix 6H Additional TRACG Outputs and Parametrics Cases

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

Appendix 6I Results of the Containment Design Basis Calculations With Suppression Pool Bypass Leakage Assumption of 1 cm^2 (1.08E-03 ft^2)

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

Chapter 7 Instrumentation and Control Systems

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

Chapter 8 Electric Power

8.1 Introduction

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

8.1.2.1 Utility Power Grid Description

Add the following to the end of the first paragraph.

NAPS SUP 8.1-1

The output of Unit 3 is delivered to a main 500/230 kV switchyard through the unit main step-up transformers, and an intermediate switchyard as described in Sections 8.2 and 8.3. The main switchyard serves four 500 kV lines and one 230 kV line. The plant is connected to the main switchyard by a 500 kV normal preferred transmission line, and a 230 kV alternate preferred transmission line that supplies power to the two reserve auxiliary transformers. The 500 kV lines go to the Ladysmith, Morrisville, and Midlothian substations. The 230 kV line goes to the Gordonsville substation. These intra-system ties transit from the NAPS main switchyard to the east, west, north, and south as shown in Figure 8.2-203. Dominion's transmission system and intra-system ties are further described in Section 8.2.

8.2 Offsite Power Systems

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

8.2.1.1 Transmission System

Replace this section with the following.

NAPS COL 8.2.4-1-A

NAPS, that is, Units 1, 2 and 3, is connected to the Dominion transmission system by four 500 kV lines (three of which were constructed for Units 1 and 2) and one 230 kV line. The lines are designed and located to minimize the likelihood of simultaneous failure.

The Unit 3 main generator feeds electric power through a 27 kV isolated-phase bus to a bank of three single-phase transformers, stepping the generator voltage up to the transmission voltage of 500 kV. Figure 8.2-201 provides a one-line diagram of the electric system from the switchyard to the onsite system. The physical arrangement of power lines from offsite power sources is shown in Figure 8.2-202. Figure 8.2-203 maps the offsite transmission lines.

The transmission lines and towers connecting the switchyard to the transmission system are as follows:

- Two 500 kV overhead lines to the Ladysmith substation (approximately 15 miles)
- A 500 kV overhead line to the Midlothian substation (approximately 41 miles)
- A 500 kV overhead line to the Morrisville substation (approximately 33 miles)
- A 230 kV overhead line to the Gordonsville substation (approximately 31 miles)

The two Ladysmith lines (one of which was constructed for Units 1 and 2) utilize a common right-of-way. Each of the other lines utilizes separate rights-of-way. The 230 kV Gordonsville line crosses under the 500 kV Ladysmith and Morrisville lines near the switchyard.

Transmission tower separation, line installation, and clearances are consistent with the National Electric Safety Code (NESC) and Dominion transmission line standards. Basic tower structural design parameters, including the number of conductors, height, materials, color, and finish are consistent with Dominion transmission line design standards. Adequate clearance exists between wire galloping ellipses to minimize conductor or structure damage. (Reference 8.2-202)

8.2.1.2 Offsite Power System

Replace the first and second paragraphs with the following.

NAPS COL 8.2.4-3-A
NAPS COL 8.2.4-4-A

The offsite power system is a nonsafety-related system. Power is supplied to the plant from multiple independent and physically separate offsite power sources. The normal preferred power source is any one of the four 500 kV lines, and the alternate preferred power source is any other one of the four 500 kV lines.

The normal preferred power source is supplied to the UATs through the intermediate transformer, MODs and isolation circuit breakers. The normal preferred power interface with the offsite power system occurs at the incoming disconnect switch of the intermediate switchyard. The MOD feeding a faulted UAT will be opened after the UAT high voltage breaker opens.

Delete the last paragraph and add the following paragraph.

Underground cables connect the normal and alternate preferred power sources to the UATs and RATs, respectively. The underground cables have a metallic sheath to prevent moisture ingress into the cable insulation. The metallic sheath is machine applied to the cable core and mechanically sealed to form a continuous barrier against moisture. To maintain their independence from each other, the underground cables are routed in duct banks and are physically and electrically separate from each other. Manholes associated with these duct banks are inspected every six months for excessive accumulation of water.

Control, instrumentation, and miscellaneous power cables associated with the normal and alternate preferred circuits are routed in duct bank between the power block and the Intermediate Switchyard. Adequate separation is ensured by either routing cables associated with the normal preferred circuit in a separate duct bank from cables associated with the alternate preferred circuit, or by routing these cables in separate conduits within the same duct bank.

8.2.1.2.1 **Switchyard**

Replace the last paragraph with the following.

NAPS COL 8.2.4-2-A
NAPS COL 8.2.4-6-A
NAPS COL 8.2.4-7-A
NAPS COL 8.2.4-8-A

The NAPS switchyard, prior to the point of interconnection with Unit 3, is a 500/230 kV, air-insulated, breaker-and-a-half bus arrangement. Unit 3 is connected to this switchyard by an overhead conductor circuit.

The physical location and electrical interconnection of the switchyard is shown on Figure 8.2-201 and Figure 8.2-202.

Control and relay protection systems are provided. Support systems, such as grounding, raceway, lighting, AC/DC station service, and switchyard lightning protection, are also provided.

The North Anna switchyard uses surge suppressors on the high and low sides of Transformers 1, 2, 3, 5, and 6. The insulation coordination and surge protective devices are applied in compliance with IEEE 1313.2 2004, “IEEE Guide for the Application of Insulation Coordination,” and IEEE C62.22 2003, “IEEE Guide for the Application of Metal Oxide Surge Arrester for Alternating Current Systems.” The surge protective devices are maintained according to NEMA requirements and manufacturer's recommendations.

A shield wire arrangement is designed for lightning abatement in the switchyard in accordance with IEEE 62.22 2003, "IEEE Guide for the Application of Metal Oxide Surge Arrestors for Alternating Current Systems," IEEE 988-2000, "Guide to Direct Lightning Shielding of Substations," and "Insulation Coordination for Power Systems."

The capacity and electrical characteristics for switchyard equipment are as follows:

Transformers	Voltage Rating	MVA Rating
Transformer	500/230 kV	67.2/89.6/112
Transformer	500/230 kV	112/145

Breakers	Max Design (kV)	Rated Current (A)	Interrupting Current at Max kV
500 kV	550	3000	40 kAIC
230 kV	242	2000	40 kAIC

Transmission Lines	Rated Current at 100ºF
500 kV	3954A
230 kV	2190A

Bus Work	Rated Current at 100ºF
500 kV	3891A
230 kV	2750A

NAPS COL 8.2.4-5-A

8.2.1.2.2 Protective Relaying

The 500 kV transmission lines are protected with redundant high-speed relay schemes with re-closing and communication equipment to minimize line outages. The 500 kV switchyard buses have redundant bus differential protection using separate and independent current and control circuits. Generating unit tie-lines and auxiliary transformer underground cable circuits are protected with redundant high-speed relay schemes. Transformers are protected with differential and over-current relay schemes.

Dominion is responsible for engineering, constructing, operating, and maintaining its electric transmission system, and for interfacing with PJM, the Regional Transmission Organization (RTO). Dominion's responsibility includes designing, maintaining, and operating all switchyard protective relaying associated with connecting Unit 3 to the North Anna switchyard. PJM studied the interconnection of Unit 3 to the North Anna switchyard and recommended no additional design requirements above those typically used by Dominion in the design of the protective relaying scheme at the switchyard.

Breakers are equipped with dual trip coils. Each redundant protection circuit that supplies a trip signal is powered from its redundant DC power load group and connected to a separate trip coil. Equipment and cabling associated with each redundant system is physically separated from its redundant counterpart. Breakers are provided with a breaker failure scheme that isolates a breaker that fails to trip due to a malfunction.

NAPS SUP 8.2-2

8.2.1.2.3 Testing and Inspection

Transmission lines are inspected via an aerial inspection program approximately twice per year. The inspection focuses on such items as right-of-way encroachment, vegetation management, conductor and line hardware condition, and the condition of supporting structures.

Routine switchyard inspection activities include, but are not necessarily limited to, the following:

- Daily transformer inspections
- Periodic inspections of circuit breakers and batteries
- Quarterly infrared scans
- Semi-annual infrared scans (relay panels)
- Semi-annual inspection of substation equipment
- Annual infrared scans
- Annual corona camera scan

Routine switchyard testing activities include, but are not necessarily limited to, the following:

- Semiannual dissolved gas analysis on transformers
- Biennial circuit breaker profile or timing tests
- Biennial 500 kV relay testing

- Triennial 230 kV relay testing
- 4-year dissolved gas analysis on transformer load tap changers
- 5-year battery discharge testing
- 8-year PT testing
- 8-year ground grid testing
- 10-year CCVT testing
- 10-year arrester testing
- 10-year wave trap testing

Switchyard protection system monitoring, maintenance, and testing are performed in accordance with North American Electric Reliability Corporation (NERC) Standard PRC-005-1, "Transmission and Generation Protection System Maintenance and Testing," Standard PRC-008-0, "Underfrequency Load Shedding Equipment Maintenance Program," and Standard PRC-017-0, "Special Protection System Maintenance and Testing."

8.2.2.1 Reliability and Stability Analysis

Replace this section with the following.

NAPS COL 8.2.4-9-A
NAPS COL 8.2.4-10-A

A system impact study analyzed load flow, transient stability and fault analysis for the addition of Unit 3. (Reference 8.2-201) The study was prepared using 2011 summer light-load and 2014 summer base-case projections.

The analysis was performed using Power Technology International Software PSS/E. The analysis examined conditions involving loss of the largest generating unit, loss of the most critical transmission line, and multiple facility contingencies. The study also examined import/export power flows between transmission system utilities.

NAPS COL 8.2.4-10-A

The equipment considered is from the point of interconnection of Unit 3 to the switchyard out to the 500 kV transmission system. This included the 230 kV buses and interconnections. The 34.5 kV portion of the North Anna switchyard is not modeled separately, but the 34.5 kV loads are considered at the 500 kV level. Maximum and minimum switchyard voltage limits have been established for the 500 kV switchyard at 534 kV and 505 kV, respectively. Normal operating and abnormal procedures exist to maintain the switchyard voltage schedule and address challenges to the maximum and minimum limits. Upon approaching or exceeding a limit, these procedures verify the availability of required and contingency equipment and materials, and direct notifications to outside agencies, until the normal voltage schedule can be maintained. Dominion has established a Switchyard Interface Agreement and protocols for Maintenance, Communications, Switchyard Control, and System Analysis sufficient to safely operate and maintain the power station interconnection to the transmission system.

The TSO provides analysis capabilities for both Long Term Planning and Real Time Operations. System conditions are evaluated to ensure a bounding analysis and model parameters are selected that are influential in determining the system's ability to provide offsite power adequacy. Elements included in the analysis are system load forecasts (including sufficient margin to ensure a bounding analysis over the life of the study), system generator dispatch (including outages of generators known to be particularly influential in offsite power adequacy of affected nuclear units), outage schedules for transmission elements that have significant influence on offsite power adequacy, cross-system power transfers and power imports/exports, and system modification plans and schedules. A Real Time State Estimator is used to assist in the evaluation of actual system conditions. These capabilities are described in the System Analysis Protocol of the Switchyard Interface Agreement.

The study concluded that with the additional generating capacity of Unit 3, the transmission system remains stable under the analyzed conditions, preserving the grid connection and supporting the normal and shutdown power requirements of Unit 3.

The reliability of the overall system design is indicated by the fact that there have been no widespread system interruptions. Failure rates of individual facilities are low. Transmission lines are designed to have less than one lightning flashover per 100 miles per year, and the record shows much better performance, indicating conservative designs. Most lightning-caused outages are momentary, with few instances of line damage. Other facilities do fail occasionally, but these are random occurrences, and experience has shown that equipment specifications are adequate.

Grid availability in the region over the past 20 years was also examined and it was confirmed that the system has been highly reliable with minimal outages due to equipment failures.

Grid stability is evaluated on an ongoing basis based on load growth, the addition of new transmission lines, or new generation capacity.

NAPS SUP 8.2-3

8.2.2.3 Failure Modes and Effects Analysis

8.2.2.3.1 Introduction

There are no single failures that can prevent the NAPS offsite power system from performing its function to provide power to Unit 3. (Reference 8.2-201)

8.2.2.3.2 Transmission System Evaluation

Unit 3 is connected to the Dominion transmission system via four 500 kV and one 230 kV overhead transmission lines. The normal preferred power source is any one of the four 500 kV lines. (See Section 8.2.1.1 and Section 8.2.1.2.)

Each transmission line occupies a separate right-of-way, except the two parallel Ladysmith lines, which share the same right-of-way. The 500 kV towers provide clearances consistent with the NESC. The towers are grounded with either ground rods or a counterpoise ground system. Failure of any one tower due to structural failure can at most disrupt and cause a loss of power distribution to itself and the adjacent line.

Failure of a line conductor would cause the loss of one of the four 500 kV lines, with the other three lines remaining available as normal and alternate preferred power sources.

8.2.2.3.3 Switchyard Evaluation

A breaker-and a-half scheme is incorporated in the design of the switchyard. The equipment in the switchyard is rated and positioned within the bus configuration according to the following criteria in order to maintain incoming and outgoing load flow from Unit 3.

- Equipment continuous current ratings are such that no single contingency in the switchyard (e.g., a breaker being out of service for maintenance) results in current exceeding 100 percent of the continuous current rating of the equipment.
- Interrupting duties are such that no faults occurring on the system exceed the equipment rating.
- Momentary ratings are such that no fault occurring on the system exceeds the equipment momentary rating.

- Voltage ratings for the equipment are specified to be greater than the maximum expected operating voltage.

The breaker-and-a-half switchyard arrangement offers the following flexibility to control a failed condition within the switchyard:

- Any faulted transmission line into the switchyard can be isolated without affecting any other transmission line.
- Either bus can be isolated without interruption of any transmission line or other bus.
- All relay schemes used for protection of the offsite power circuits and the switchyard equipment include primary and backup protection features. All breakers are equipped with dual trip coils. Each protection circuit that supplies a trip signal is connected to a separate trip coil.

8.2.2.3.4 **Intermediate Switchyard**

The intermediate switchyard is an integral part of the normal preferred power supply. The failure of any component within the intermediate switchyard may disrupt the normal preferred power supply. However, the alternate preferred power supply will remain available to supply the load.

The equipment in the intermediate switchyard is rated according to the following criteria:

- Interrupting duties are specified such that no faults occurring on the system exceed the equipment rating.
- Momentary ratings are specified such that no faults occurring on the system exceed the equipment momentary rating.
- Voltage ratings are specified to be greater than the maximum expected operating voltage.
- Circuit breaker continuous current ratings are chosen such that no single contingency will result in a load exceeding 100 percent of the nameplate continuous current rating of the breaker.

The normal preferred and alternate preferred power supplies are electrically independent and are physically separate from each other.

Therefore, a minimum of one preferred source of power remains available to supply the load during all plant conditions.

8.2.3 Design Basis Requirements

STD COL 8.2.4-9-A Delete the parenthetical statement at the end of the ninth bullet-list entry.

8.2.4 COL Information

8.2.4-1-A Transmission System Description

NAPS COL 8.2.4-1-A This COL item is addressed in Section 8.2.1.1.

8.2.4-2-A Switchyard Description

NAPS COL 8.2.4-2-A This COL item is addressed in Section 8.2.1.2.1.

8.2.4-3-A Normal Preferred Power

NAPS COL 8.2.4-3-A This COL item is addressed in Section 8.2.1.2.

8.2.4-4-A Alternate Preferred Power

NAPS COL 8.2.4-4-A This COL item is addressed in Section 8.2.1.2.

8.2.4-5-A Protective Relaying

NAPS COL 8.2.4-5-A This COL item is addressed in Section 8.2.1.2.2.

8.2.4-6-A Switchyard DC Power

NAPS COL 8.2.4-6-A This COL item is addressed in Section 8.2.1.2.1.

8.2.4-7-A Switchyard AC Power

NAPS COL 8.2.4-7-A This COL item is addressed in Section 8.2.1.2.1.

8.2.4-8-A Switchyard Transformer Protection

NAPS COL 8.2.4-8-A This COL item is addressed in Section 8.2.1.2.1.

8.2.4-9-A Stability and Reliability of the Offsite Transmission Power Systems

NAPS COL 8.2.4-9-A This COL item is addressed in Section 8.2.2.1.

8.2.4-10-A Interface Requirements

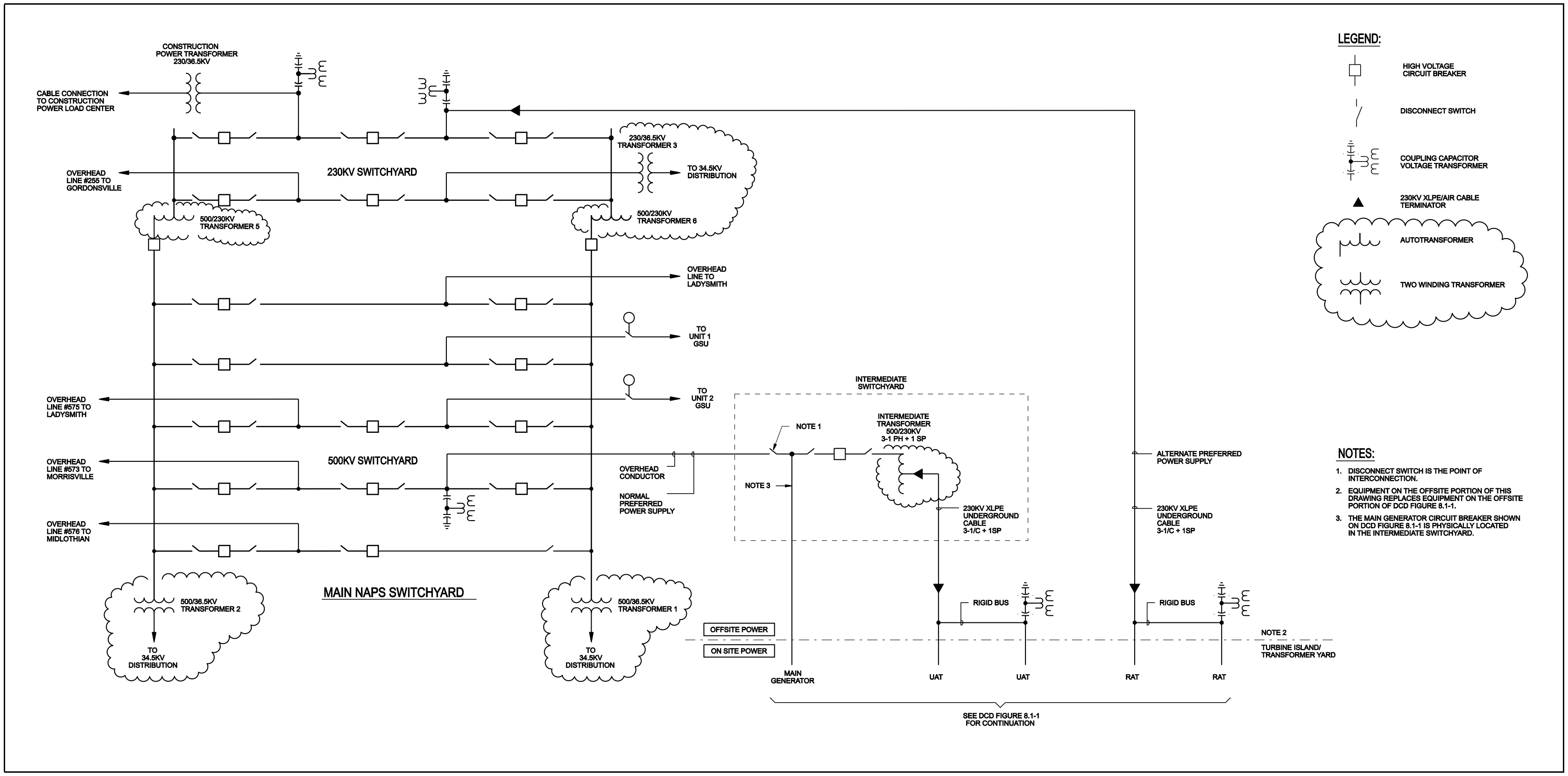
NAPS COL 8.2.4-10-A This COL item is addressed in Section 8.2.2.1.

8.2.5 References

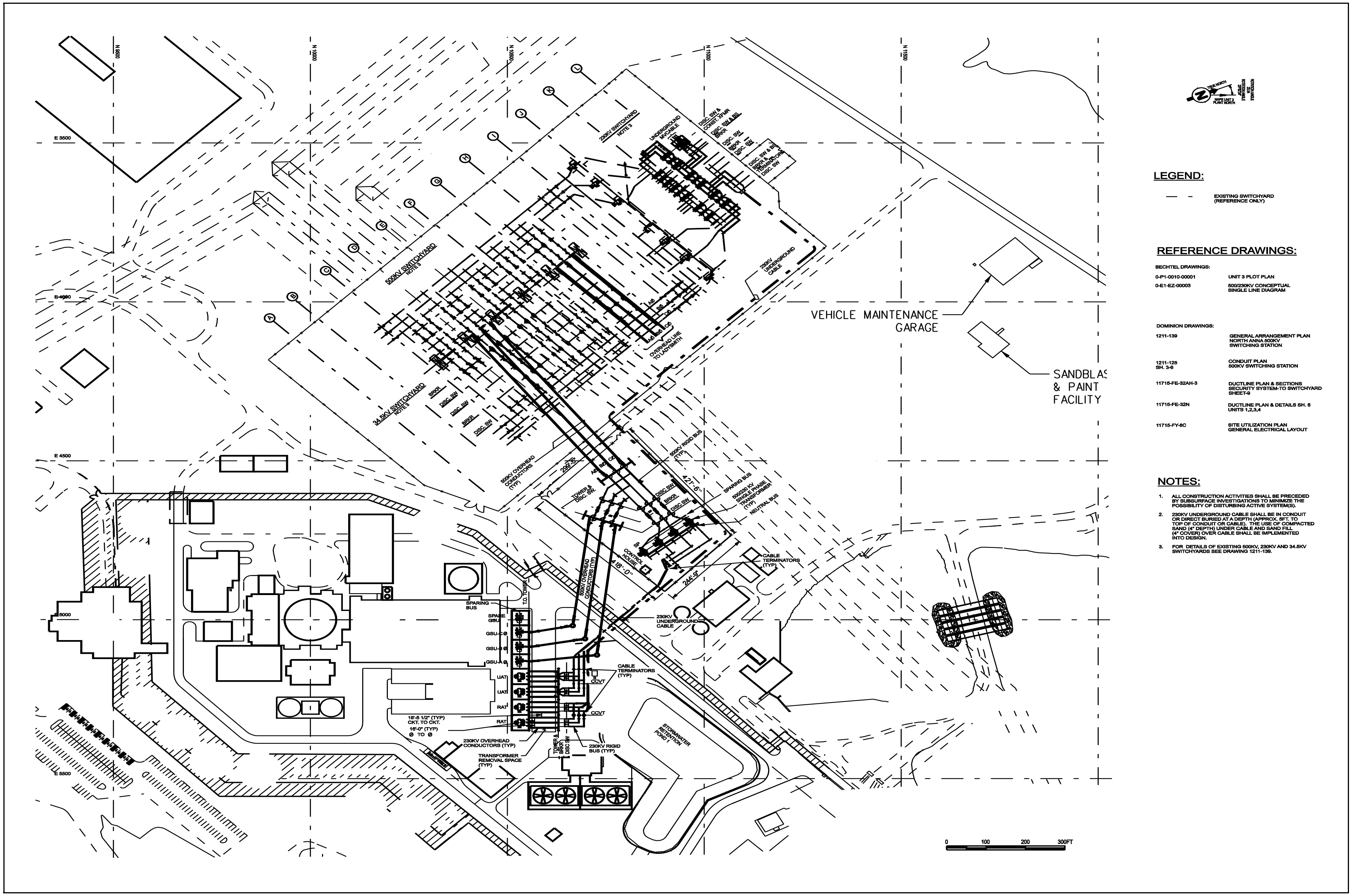
8.2-201 PJM Generator Interconnection Q65 North Anna 500 kV System Impact Study, June 2007.

8.2-202 VA PJM Design and Application of Overhead Transmission Lines 69kV and above, May 20, 2002.

NAPS COL 8.2.4-1-A Figure 8.2-201 500/230 kV Switchyard Single-Line Diagram

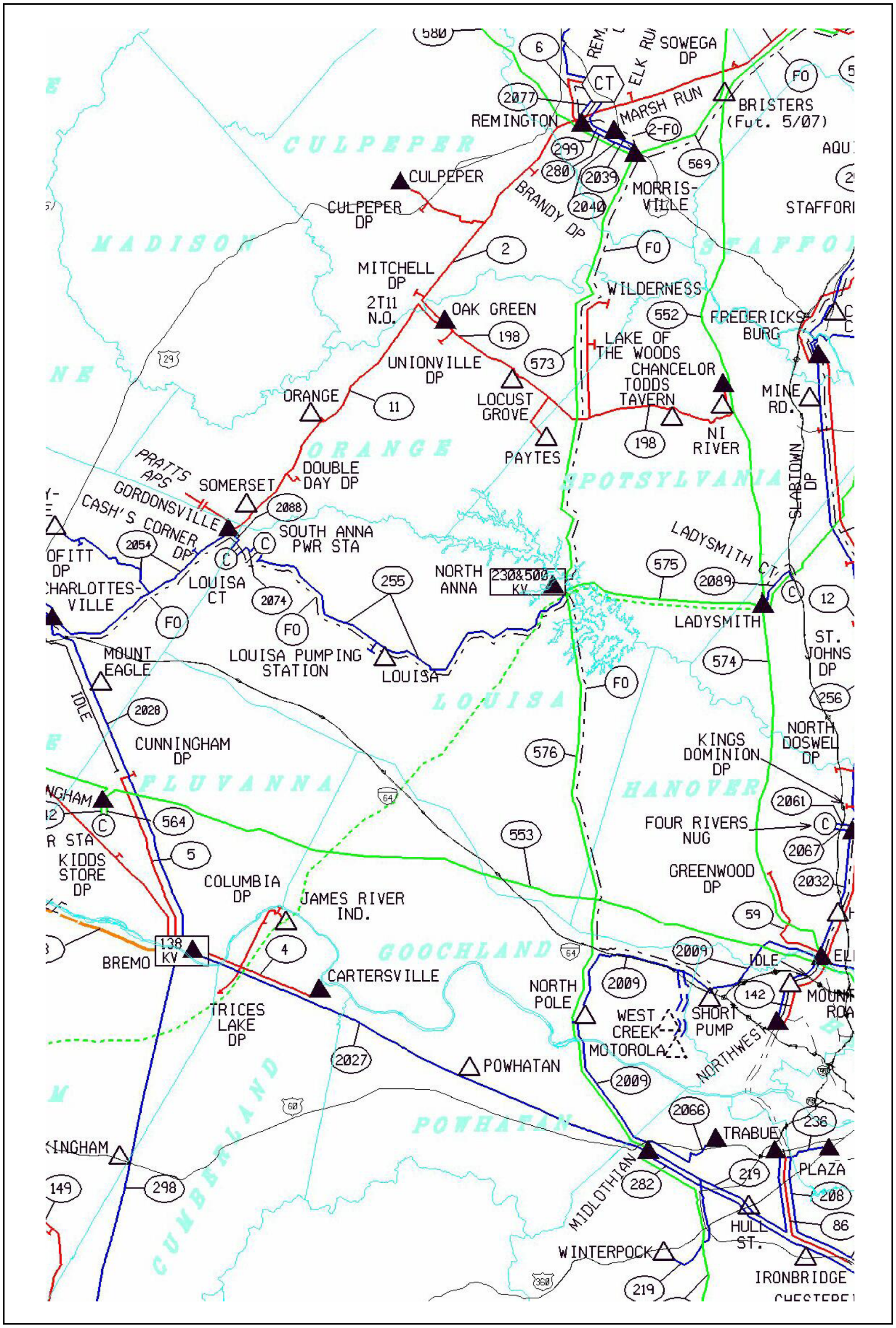


NAPS COL 8.2.4-1-A **Figure 8.2-202 500/230 kV Switchyard Arrangement**



NAPS SUP 8.1-1

Figure 8.2-203 Dominion Transmission Line Map



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 Description

Insert the following as the first paragraph.

NAPS SUP 8.3-1

An intermediate switchyard is utilized to transition off-site power from the NAPS switchyard to the Unit 3 main power transformers, and unit auxiliary transformers (UATs). This intermediate switchyard contains the main generator circuit breaker, and a supply circuit breaker, which provides power to 500/230 kV intermediate transformers used to supply power to the UATs. These intermediate transformers consist of three single phase transformers and include an installed spare transformer. Also included in the intermediate switchyard is a transmission tower which supports a 500 kV disconnect switch that is identified as the point of interconnection between the onsite power sources and the offsite power sources. This point of interconnection is the demarcation between Unit 3 and the NAPS switchyard and transmission system. (See Figure 8.2-201)

8.3.2.1.1 Safety-Related Station Batteries and Battery Chargers

Station Blackout

Add the following paragraph at the end of this section.

NAPS SUP 8.3-2

Training and procedures to mitigate an SBO event are implemented in accordance with Sections 13.2 and 13.5, respectively. As recommended by NUMARC 87-00 (Reference 8.3-201), SBO event mitigation procedures address SBO response (e.g., restoration of on-site standby power sources), AC power restoration (e.g., coordination with transmission system load dispatcher), and severe weather guidance (e.g., identification of site-specific actions to prepare for the onset of severe weather such as an impending tornado), as applicable. The ESBWR 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 in DCD Section 15.5.5, Station Blackout. In addition, there are no nearby large power sources, such as a gas turbine or black start fossil fuel plant, that can directly connect to the station to mitigate the SBO event.

Restoration from an SBO event will be contingent upon power being made available from any one of the following sources:

- Any of the standby or ancillary diesel generators.
- Restoration of any one of the four 500 kV transmission lines described in Section 8.2.
- Restoration of the 230 kV transmission line described in Section 8.2.

8.3.5 References

8.3-201 Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors, NUMARC 87-00, Revision 1, August 1991.

Appendix 8A Miscellaneous Electrical Systems

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

8A.2.1 Description

Replace DCD Section 8A.2.1 with the following.

NAPS COL 8A.2.3-1-A A cathodic protection system is provided to the extent required. The system is designed in accordance with the requirements of the National Association of Corrosion Engineers (NACE) Standards (DCD Reference 8A-5).

8A.2.3 COL Information

8A.2.3-1-A Cathodic Protection System

NAPS COL 8A.2.3-1-A This COL item is addressed in Section 8A.2.1.

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.

9.1.1.7 Safety Evaluation

Structural Design

STD COL 9.1-4-A Delete the last sentence of the third paragraph.

Protection Features of the New Fuel Storage Facilities

STD COL 9.1-4-A Delete the last sentence of the third paragraph.

9.1.4 Light Load Handling System (Related to Refueling)

9.1.4.13 Refueling Operations

Add the following at the end of this section.

STD COL 9.1-4-A Section 13.5 requires development of fuel handling procedures. Fuel handling procedures address the status of plant systems required for refueling; inspection of replacement fuel and control rods; designation of proper tools; proper conditions for spent fuel movement and storage; proper conditions to prevent inadvertent criticality; proper conditions for fuel cask loading and movement; and status of interlocks, reactor trip circuits and mode switches. These procedures provide instructions for use of refueling equipment, actions for core alterations, monitoring core criticality status, and accountability of fuel for refueling operations. Fuel handling procedures are developed six months before fuel receipt to allow sufficient time for plant staff familiarization, to allow NRC staff adequate time to review the procedures, and to develop operator licensing examinations.

Personnel qualifications and training for fuel handlers are addressed in Section 13.2.

9.1.4.19 Inspection and Testing Requirements

Add the following at the end of this section.

STD COL 9.1-4-A Section 17.5 describes the QA program that is applied to monitoring, implementing, and ensuring compliance with fuel handling procedures.

As part of normal plant operations, the fuel-handling equipment is inspected for operating conditions before each refueling operation. During the operational testing of this equipment, procedures are followed that will affirm the correct performance of the fuel-handling system interlocks. Other maintenance and test procedures are developed based on manufacturer's requirements.

9.1.5 Overhead Heavy Load Handling Systems (OHLHS)

9.1.5.6 Other Overhead Load Handling System

Add the following at the end of this section.

STD COL 9.1-5-A

Special Lifting Devices

Testing and inspection of special lifting devices follow the guidelines of ANSI N14.6.

Other Lifting Devices

Slings used for heavy load lifts meet the requirements specified for slings in ANSI B30.9 and the guidance specified in NUREG-0612, Section 5.1.1(5).

9.1.5.8 Operational Responsibilities

Replace this section with the following.

STD COL 9.1-5-A

Procedures

Section 13.5 requires the development of administrative procedures to control heavy loads prior to fuel load to allow sufficient time for plant staff familiarization, to allow NRC staff adequate time to review the procedures, and to develop operator licensing examinations. Heavy loads handling procedures address:

- Equipment identification
- Required equipment inspections and acceptance criteria prior to performing lift and movement operations
- Approved safe load paths and exclusion areas
- Safety precautions and limitations
- Special tools, rigging hardware, and equipment required for the heavy load lift
- Rigging arrangement for the load

- Adequate job steps and proper sequence for handling the load

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 or spent fuel pool or on safe shutdown equipment. Paths are defined in procedures and equipment layout drawings. Safe load path procedures address the following general requirements:

- When heavy loads must be carried directly over the spent fuel pool, reactor vessel or safe shutdown equipment, procedures will limit the height of the load and the time the load is carried.
- When heavy loads could be carried (i.e., no physical means to prevent) but are not required to be carried directly over the spent fuel pool, reactor vessel or safe shutdown equipment, procedures will define an area over which loads shall not be carried so that if the load is dropped, it will not result in damage to spent fuel or operable safe shutdown equipment or compromise reactor vessel integrity.
- Where intervening structures are shown to provide protection, no load travel path is required.
- Defined safe load paths will follow, to the extent practical, structural floor members.
- When heavy loads movement is restricted by design or operational limitation, no safe load path is required.
- Supervision is present during heavy load lifts to enforce procedural requirements.

Inspection and Testing

Cranes addressed in this section are inspected, tested, and maintained in accordance with Section 2-2 of ANSI B30.2, Section 11.2 of ANSI B30.11, or Sections 16-1.2.1 and 16-1.2.3 of ANSI B30.16 with the exception that tests and inspections may be performed prior to use for infrequently used cranes. Prior to making a heavy load lift, an inspection of the crane is made in accordance with the above applicable standards.

Training and Qualification

Training and qualification of operators of cranes addressed in this section meet the requirements of ANSI B30.2, and include the following:

- Knowledge testing of the crane to be operated in accordance with the applicable ANSI crane standard.

- Practical testing for the type of crane to be operated.
- Supervisor signatory authority on the practical operating examination.
- Applicable physical requirements for crane operators as defined in the applicable crane standard.

Quality Assurance

Procedures for control of heavy loads are developed in accordance with Section 13.5. In accordance with Section 17.5, other specific quality program controls are applied to the heavy loads handling program, targeted at those characteristics or critical attributes that render the equipment a significant contributor to plant safety.

9.1.5.9 Safety Evaluations

Add the following at the end of this section.

STD COL 9.1-5-A No heavy loads are identified that are outside the scope of the certified design. In addition, there is no heavy load handling equipment, nor interlocks associated with heavy load handling equipment, outside the scope of the certified design.

9.1.6 COL Information

9.1-4-A Fuel Handling Operations

STD COL 9.1-4-A This COL item is addressed in Section 9.1.4.13 and Section 9.1.4.19.

9.1-5-A Handling of Heavy Loads

STD COL 9.1-5-A This COL item is addressed in Section 9.1.5.6, Section 9.1.5.8, and Section 9.1.5.9.

9.2 Water Systems

9.2.1 Plant Service Water System

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

9.2.1.2 System Description

Replace the Summary Description, Detailed System Description, and Operation portions of this section with the following.

NAPS CDI

Summary Description

The PSWS rejects heat from nonsafety-related RCCWS and TCCWS heat exchangers to the environment. The source of cooling water to the PSWS is from the auxiliary heat sink (AHS), while the heat removed is rejected to the AHS. Unit 3 utilizes mechanical draft plume abated cooling towers for the AHS.

A simplified diagram of the PSWS is shown in Figure 9.2-201.

Detailed System Description

The PSWS consists of two independent and 100 percent redundant trains that continuously circulate water through the RCCWS and TCCWS heat exchangers.

Each PSWS train consists of two 50 percent capacity vertical pumps taking suction in parallel from the plant service water basin. Discharge is through a check valve, a self-cleaning strainer, and a motor-operated discharge valve at each pump to a common header. Each common header supplies plant service water to each RCCWS and TCCWS heat exchanger train arranged in parallel. The plant service water is returned via a common header to the mechanical draft plume abated cooling tower (AHS) in each train. Remote-operated isolation valves and a cross-tie line permit routing of the plant service water to either cooling tower. The RCCWS and TCCWS heat exchangers are provided with remotely-operated isolation valves. Flow control valves are provided at each heat exchanger outlet.

The PSWS pumps are located at the plant service water basin. Each pump is sized for 50 percent of the train flow requirement for normal operation. The pumps are low speed, vertical wet–pit designs with allowance for increase in system friction loss and impeller wear. The design of the heat rejection facilities and PSWS pumps have sufficient available net positive suction head (NPSH) under worst case conditions. Basin water level is monitored to ensure sufficient NPSH at design flow is provided to the PSWS pumps.

The pumps in each train are powered from redundant electrical buses. During a LOPP, the pumps are powered from the two nonsafety-related standby diesel-generators.

Where needed, valves are provided with hard seats to withstand erosion. The valves are arranged for ease of maintenance, repair, and in-service inspection. During a LOPP, the motor-operated valves are powered from the two nonsafety-related standby diesel-generators.

The AHS provided for each PSWS train is a separate, multi-celled, 100 percent capacity mechanical draft plume abated cooling tower, with the fans in the tower from each train supplied by one of the two redundant electrical buses. During a LOPP, the fans are powered from the two nonsafety-related standby diesel-generators. Each tower cell has an adjustable-speed, reversible motor fan unit that can be controlled for cold weather conditions to prevent freezing in the basin. A full flow bypass is provided to return water directly to the PSWS basin to allow ease of cold weather startup. Mechanical and electrical isolation allows maintenance on one tower, including complete disassembly, during full power operation. The Station Water System (SWS) provides makeup for blowdown, drift, and evaporation losses from the basin. Refer to Section 9.2.10 for the SWS discussion. Fiberglass reinforced polyester pipe is used for buried PSWS piping to preclude long-term corrosion.

Replace the eighth sentence in the sixth paragraph with the following.

NAPS COL 9.2.1-1-A Fiberglass reinforced polyester pipe is used for buried PSWS piping to preclude long-term corrosion. Appropriate chemical treatment is added to the PSWS basin to preclude long-term corrosion and fouling of the PSWS components based on site water quality analysis. PSWS materials are compatible with the PSWS water treatment regime.

In the event of a LOPP, the PSWS supports the RCCWS in bringing the plant to cold shutdown condition in 36 hours assuming the most limiting single active or passive component failure.

Unit 3 PSWS heat loads are shown in DCD Table 9.2-1. The PSWS component design characteristics are shown in Table 9.2-201.

The PSWS design detects and alarms in the MCR any potential gross leakage and permits the isolation of any such leak in a sufficiently short period of time to preclude extensive plant damage.

Analysis of routine PSWS basin grab samples will detect RCCWS leakage, which may contain low levels of radioactivity, into the PSWS. This provides the action required by NRC Inspection and Enforcement Bulletin No. 80-10.

The potential for water hammer is mitigated through the use of various system design and layout features, such as automatic air release/vacuum valves installed at high points in system piping and at the pump discharge, proper valve actuation times to minimize water hammer, limiting fluid velocities in piping, procedural requirements ensuring proper line filling prior to system operation and after maintenance operations, and the use of check valves at pump discharges to prevent backflow into the pumps.

Operation

The PSWS operates during startup, normal power operation, hot standby, cooldown, shutdown/refueling, and LOPP.

During normal plant operation, the cross-tie valves in the PSWS pump discharge header are open, allowing two of the four 50 percent capacity PSWS pumps to supply water to both PSWS trains. Heat removed from the RCCWS and TCCWS is rejected to the auxiliary heat sink.

Operation of any two of the four PSWS pumps is sufficient for the design heat load removal in any normal operating mode. During normal and LOPP cooldown mode, three pumps can be used for operational convenience to bring the plant to cold shutdown condition in 24 hours.

During a LOPP, running PSWS pumps restart automatically using power supplied by the nonsafety-related standby diesel-generators.

9.2.1.6 COL Information

9.2.1-1-A Material Selection

NAPS COL 9.2.1-1-A This COL item is addressed in Section 9.2.1.2.

9.2.2 Reactor Component Cooling Water System

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

9.2.3 Makeup Water System

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

9.2.3.2 System Description

Replace the introductory text and the Demineralization Subsystem portions of this section with the following.

NAPS CDI

The MWS consists of two subsystems: 1) the demineralization subsystem and 2) the storage and transfer subsystem. The makeup water transfer pumps and the demineralization subsystem are sized to meet the demineralized water needs of all operational conditions except for shutdown/refueling/startup. During the shutdown/refueling/startup mode, the increases in plant water consumption require use of a temporary demineralization subsystem and temporary makeup water transfer pumps to be used as a supplemental water source.

The MWS major equipment is housed entirely in the Water Treatment Building except for the demineralized water storage tank (which is outdoors and adjacent to this building) and the distribution piping to the interface systems. Freeze protection is provided for the demineralized water storage tank and piping exposed to freezing conditions.

The MWS equipment and associated piping in contact with demineralized water are fabricated from corrosion resistant materials such as stainless steel to prevent contamination of the makeup water.

Table 9.2-202 lists the major MWS components.

Demineralization Subsystem

Feedwater for the demineralization subsystem is provided by the SWS. Production of demineralized water by the demineralization subsystem can be initiated and shut down either automatically (based on the demineralized water storage tank level) or manually. Feedwater is treated in the following sequence:

1. Activated carbon filters
2. Reverse osmosis modules
3. Mixed bed demineralizers

Each reverse osmosis (RO) module includes cartridge filters. The RO modules are separated by an inter-stage break tank. Chemical addition is provided upstream of the RO module cartridge filters as required. High pressure pumps provide the pressure required for flow through the RO unit membranes. The RO unit reject flow is sent to the cooling tower blowdown facility. The RO product water is temporarily stored in an RO product water storage tank before being pumped by one of the forwarding pumps to the mixed bed demineralizer unit. Operation of the RO high-pressure pumps is interlocked with that of the forwarding pumps. The mixed bed demineralizer consists of both strong cation and anion resins in the same vessel that polishes the RO product water. The mixed bed unit effluent is monitored for water quality. This effluent is automatically recirculated to the station water storage tank until the water quality requirements are met. Makeup water is then delivered to the MWS demineralized water storage tank. The modular design of the RO unit and the mixed bed unit allows continuous demineralized water production. Cleaning, back flushing, or module removal are manual operations based on elevated differential pressure across the module or total flow through the system. No regeneration of mixed bed modules is performed on-site.

9.2.4 Potable and Sanitary Water Systems

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

Delete the first paragraph and replace the last paragraph with the following.

NAPS CDI

9.2.4.1 Design Bases

Safety Design Basis

The Potable Water System (PWS) and Sanitary Waste Discharge System (SWDS) do not perform any safety-related function. Therefore, the PWS and SWDS have no safety design bases.

Power Generation Design Basis

The PWS and SWDS are designed to provide potable water supplies and sewage collection and treatment necessary for normal plant operation and shutdown periods. The PWS provides sufficient supply and is designed to supply up to 12.6 liters per second (200 gallons per minute) of potable water during peak demand periods.

The potable water system supplies the quality of water required by the authorities having jurisdiction.

The sanitary waste discharge system is designed to produce a waste water effluent quality required by Federal, state, and local regulations and permits.

9.2.4.2 System Description

Potable Water System

The PWS consists of ground wells at various locations on site. As shown on Figure 9.2-202, for each well house there is a pump, compressor, hydro-pneumatic tank, and interconnecting piping and valves. Combined potable water volume of the hydro-pneumatic tanks is 50,000 liters (13,200 gallons). Potable water from hydro-pneumatic tanks flows to a common potable water header for supply to Unit 3 facilities. The Unit 3 PWS underground header is connected to the Unit 1 and 2 domestic water header via a normally-closed isolation valve. This cross-tie connection is provided for operational flexibility and ease of system maintenance. In addition to non-radiological areas, potable water is provided to areas where inadvertent backflow into the system could result in radiological contamination of the potable water. For those PWS branches with outlets in areas where the potential for radiological contamination exists, backflow prevention is provided through the installation of backflow preventers.

Sanitary Waste Discharge System

The sanitary waste generated by Unit 3 is collected by a network of sumps and is pumped to the Unit 3 Sewage Treatment Plant (STP). The Unit 3 STP consists of two extended aeration type packaged units, each rated for a normal capacity of 94,500 liters per day (25,000 gallons per day). The two packaged units in parallel can treat 189,000 liters per day (50,000 gallons per day) of sanitary sewage. During normal plant operation, only one of the packaged units is required, and during outages, both packaged units can be operated to serve additional demand. The effluent is discharged to the cooling tower blow down sump and subsequently drained to the WHTF.

Analysis of routine STP sludge tank grab samples will detect events that might contaminate the STP downstream of the sludge tank. This provides the action required by Inspection and Enforcement Bulletin No. 80-10. The quality of effluent meets, at a minimum, the standards established by

Federal, state, and local regulations and permits. Sewage sludge is transferred to a truck for off-site disposal. A simplified diagram of the SWDS is shown in Figure 9.2-203.

9.2.4.3 Safety Evaluation

Potable Water System

The PWS has no safety-related function and is not connected to any safety-related system or component. Failure of the system does not compromise any safety-related equipment or component and does not prevent safe shutdown of the plant. The PWS does not handle radioactive fluids. It is neither connected to, nor does it interface with any system that may contain radioactive fluids.

Sanitary Waste Discharge System

The SWDS has no safety-related function and is not connected to any safety-related system or component. Failure of the system does not compromise any safety-related equipment or component and does not prevent safe shutdown of the plant.

The SWDS is not designed to handle radioactive fluids. It is neither connected to, nor does it interface with, any system that may contain radioactive fluids. As a precautionary measure, the STP sludge tank is grab sampled on a batch basis for potential radiological contamination. In the event radioactivity is detected above predetermined limits, controls are in place to initiate treatment and prevent unmonitored, uncontrolled radioactive releases to the environment.

9.2.4.4 Testing and Inspection Requirements

The PWS and SWDS are proven operable by their use during normal plant operation.

9.2.4.5 Instrumentation Requirements

The PWS and SWDS are furnished with instrumentation that permit local and/or remote monitoring, and local control of each of the respective processes. This instrumentation includes meters, switches, indicators, pressure gauges, flow switches, transmitters, controllers, and valves as required for service, operation, and protection of plant personnel and equipment.

9.2.5 Ultimate Heat Sink

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

Replace the second to last sentence in the seventh paragraph with the following.

STD COL 9.2.5-1-H Procedures that identify and prioritize available makeup sources seven days after an accident, and provide instructions for establishing necessary connections, will be developed in accordance with the procedure development milestone in Section 13.5.

9.2.5.1 COL Information

9.2.5-1-H Post 7 day Makeup to UHS

STD COL 9.2.5-1-H This COL Item is addressed in Section 9.2.5.

9.2.6 Condensate Storage and Transfer System

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

9.2.6.2 System Description

Add the following at the end of the first paragraph.

STD SUP 9.2.6-1 Freeze protection is provided for the CS&TS.

9.2.7 Chilled Water System

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

9.2.8 Turbine Component Cooling Water System

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

9.2.9 Hot Water System

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

9.2.10 Station Water System

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

9.2.10.2 System Description

Replace the Detailed System Description portion of this section with the following.

NAPS CDI

Detailed System Description

The SWS consists of the following subsystems:

- Plant Cooling Tower Makeup System (PCTMS)
- Pretreated Water Supply System (PWSS)

The PCTMS provides makeup water to the cooling tower basins for both the PSWS (**Section 9.2.1**) and CIRC (Section 10.4). The supply of water makes up for losses resulting from evaporation, drift and blowdown from the cooling towers. In addition, the PCTMS provides makeup water to replace water used for strainer backwashes. The PCTMS consists of a water source, pumps, strainers, connecting piping, valves and instrumentation. See **Figure 9.2-204** for a simplified system diagram and **Table 9.2-203** for component design parameters for the PCTMS.

The PWSS chemically conditions and filters the water supplied to the Makeup Water System (MWS) (**Section 9.2.3**) for further treatment for use as demineralized water. The PWSS also supplies water to the Fire Protection System (FPS) (**Section 9.5.1**) for filling the primary firewater tanks. In addition, the PWSS provides PSWS cooling tower makeup as an alternate to the PCTMS. The PWSS also provides water for the strainers and filter backwashes. The PWSS consists of a water source, pumps, strainers, filters, chemical injection equipment, station water storage tank (SWST), connecting piping, valves and instrumentation. See **Figure 9.2-205** for a simplified diagram and **Table 9.2-204** for component design parameters for the PWSS.

NAPS COL 9.2.1-1-A

Table 9.2-201 PSWS Component Design Characteristics

	PSWS Pumps	
	Type	Vertical, wet-pit, centrifugal turbine
	Quantity	4
	Capacity Each	1.262 m^3/s (20,000 gpm)
	Plant Service Water System [1]	
NAPS CDI	Flow (AHS)	2.524 m^3/s (40,000 gpm)
	PSWS Cooling Towers and Basins	
NAPS CDI	Type	Mechanical draft, multi-cell, adjustable speed reversible fans, plume abated
	Quantity	2
	Heat Load Each [2]	90 MW (3.07 $\times$ 10^8 BTU/hr)
	Flow Rate (Water) Each	2.524 m^3/s (40,000 gpm)
NAPS CDI	Ambient Wet Bulb Temperature [3]	26.1°C (79°F)
	Approach Temperature	5.0°C (9°F)
	Cold Leg Temperature	31.1°C (88°F)
NAPS SUP 9.2.1-1	Basin Reserve Storage Capacity [1]	2.6 million gallons
	Strainers	
	Type	Automatic cleaning basket
	Quantity	4

1. PSWS required to remove 2.02 $\times$ 10^7 MJ (1.92 $\times$ 10^{10} BTU) for period of 7 days without active makeup.
2. Cooling tower sizing capacity including margin over system design heat loads as defined in DCD Table 9.2-1.
3. Ambient web bulb temperature includes a 0.5°C (1°F) recirculation allowance.

NAPS CDI

Table 9.2-202 Major Makeup Water System Components

Two activated carbon filter feed pumps

One activated carbon filter unit consisting of multiple modules

Four 5 micron cartridge filters

Two first pass reverse osmosis (RO) high-pressure pumps

Two second pass RO booster pumps

Two second pass RO high-pressure pumps

One RO system consisting of multiple modules

One RO break tank

One chemical treatment system that provides chemical conditioning for the RO system

One chemical cleaning system for the RO membranes

NAPS CDI

Table 9.2-203 Station Water System - Plant Cooling Tower Makeup System Component Design Parameters

Pumps	
Type	Vertical, wet pit, centrifugal turbine
Quantity	3 × 50%
Capacity each	Approximately 2,700 m^3/hr (11,888 gpm)
Strainers	
Type	Duplex, basket
Quantity	3

NAPS CDI

Table 9.2-204 Station Water System – Pretreated Water Supply System Component Design Parameters

PWSS Pumps	
Type	Vertical, wet pit, centrifugal turbine
Quantity	2 × 100%
Capacity each	Approximately 170 m^3/hr (750 gpm)
FWS Makeup Pumps	
Type	Horizontal, centrifugal
Quantity	2 × 100%
Capacity each	Approximately 170 m^3/hr (750 gpm)
Miscellaneous Users Supply Pumps	
Type	Horizontal, centrifugal
Quantity	2 × 100%
Capacity each	Approximately 25 m^3/hr (110 gpm)
Storage Tank capacity	Approximately 1,100 m^3 (290,000 gallons)
Strainers	
Type	Duplex, basket
Quantity	2
PWSS Filtration System	
Quantity	1 Lot
PWSS Chemical Injection System	
Quantity	1 Lot

Figure 9.2-201 Plant Service Water System Simplified Diagram

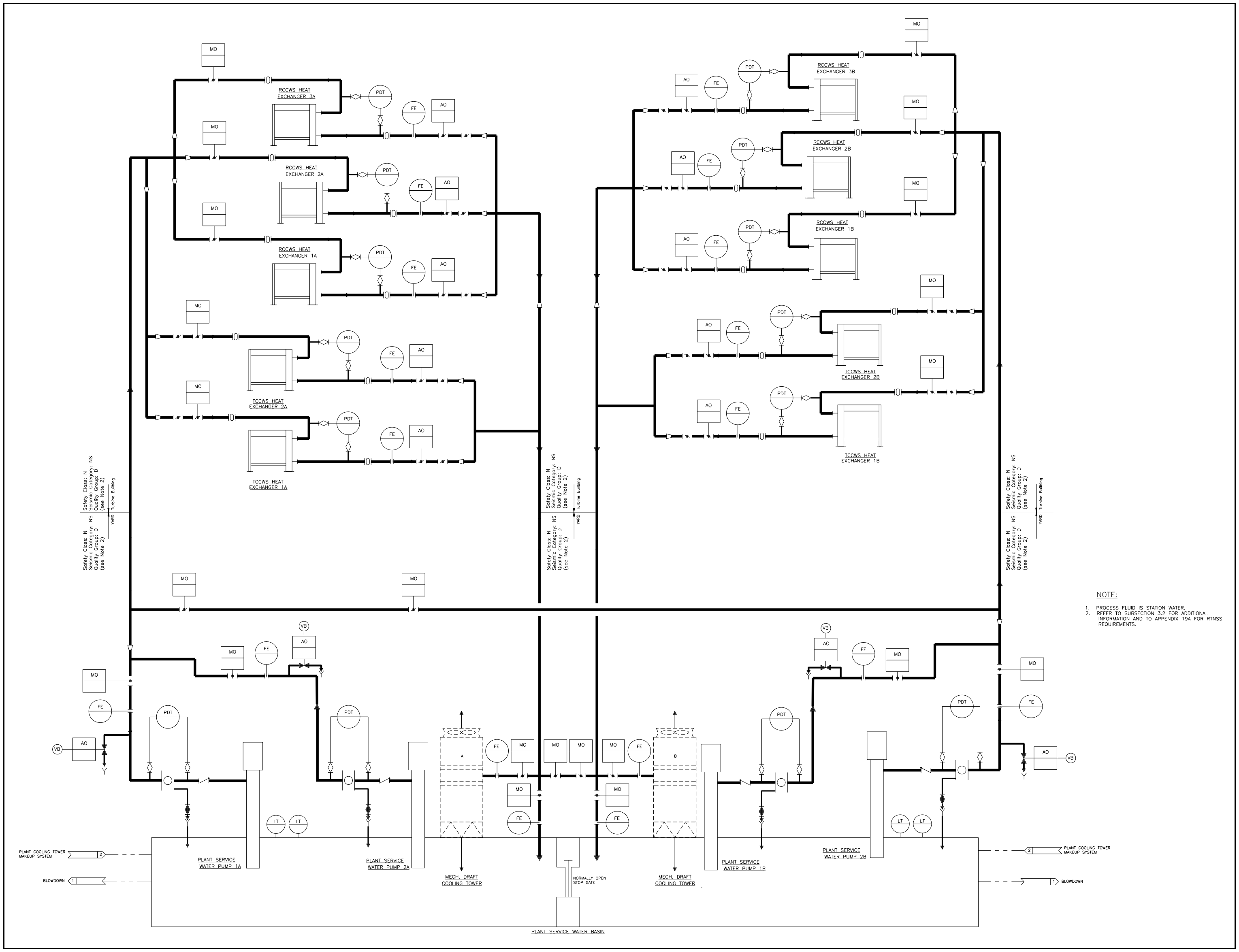


Figure 9.2-202 Potable Water System Simplified Diagram

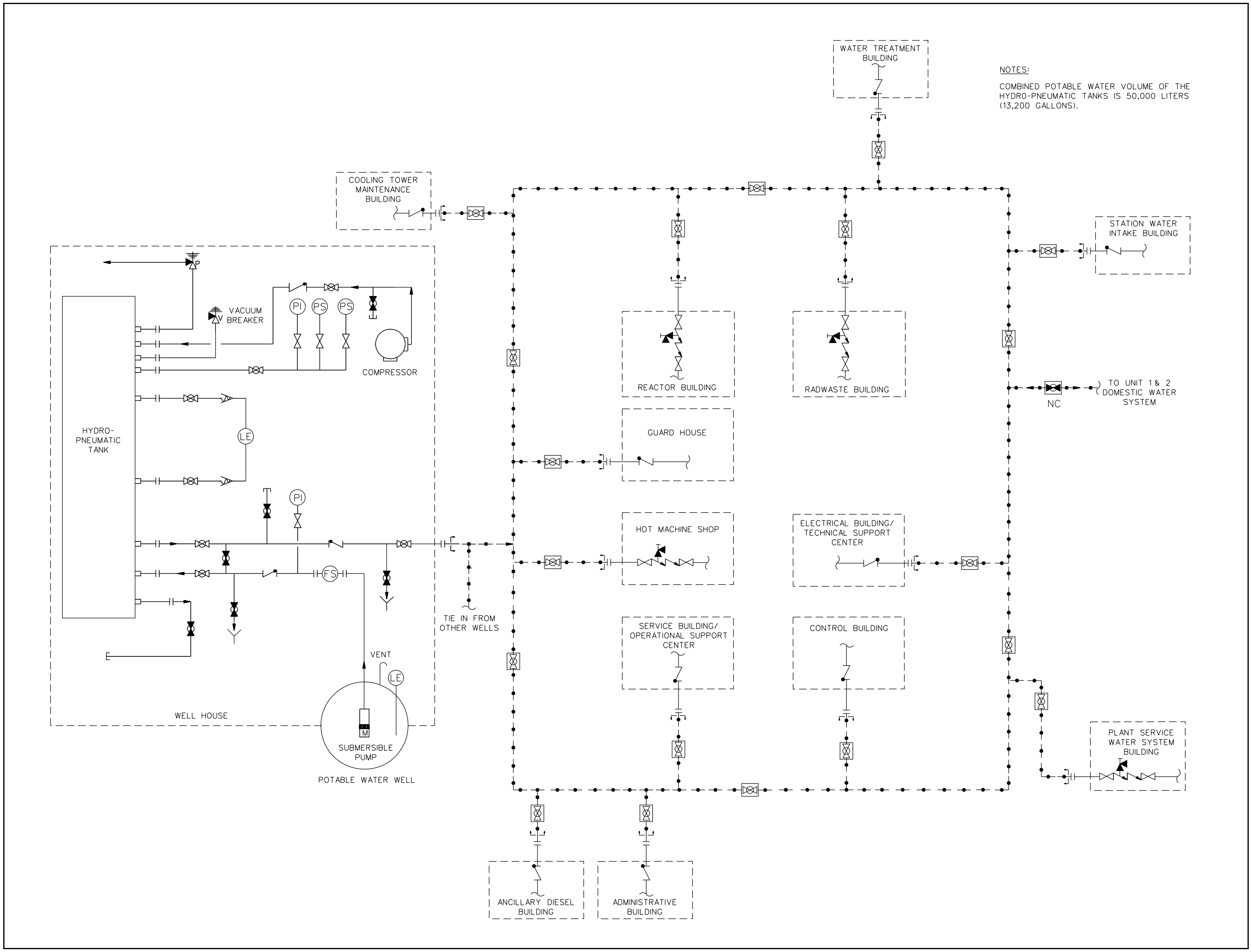


Figure 9.2-203 Sanitary Waste Discharge System Simplified Diagram

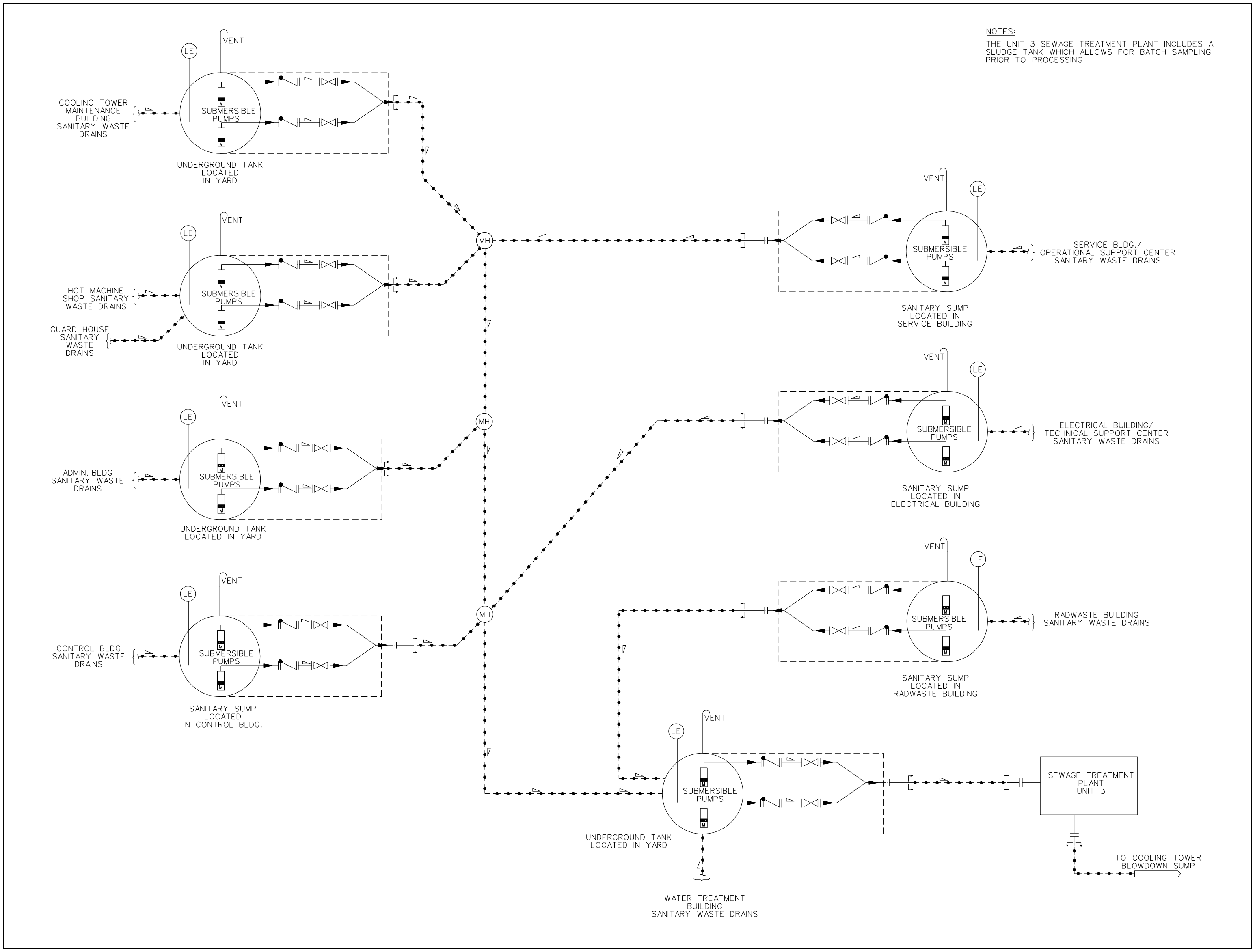


Figure 9.2-204 Station Water System - Plant Cooling Tower Makeup System (PCTMS)

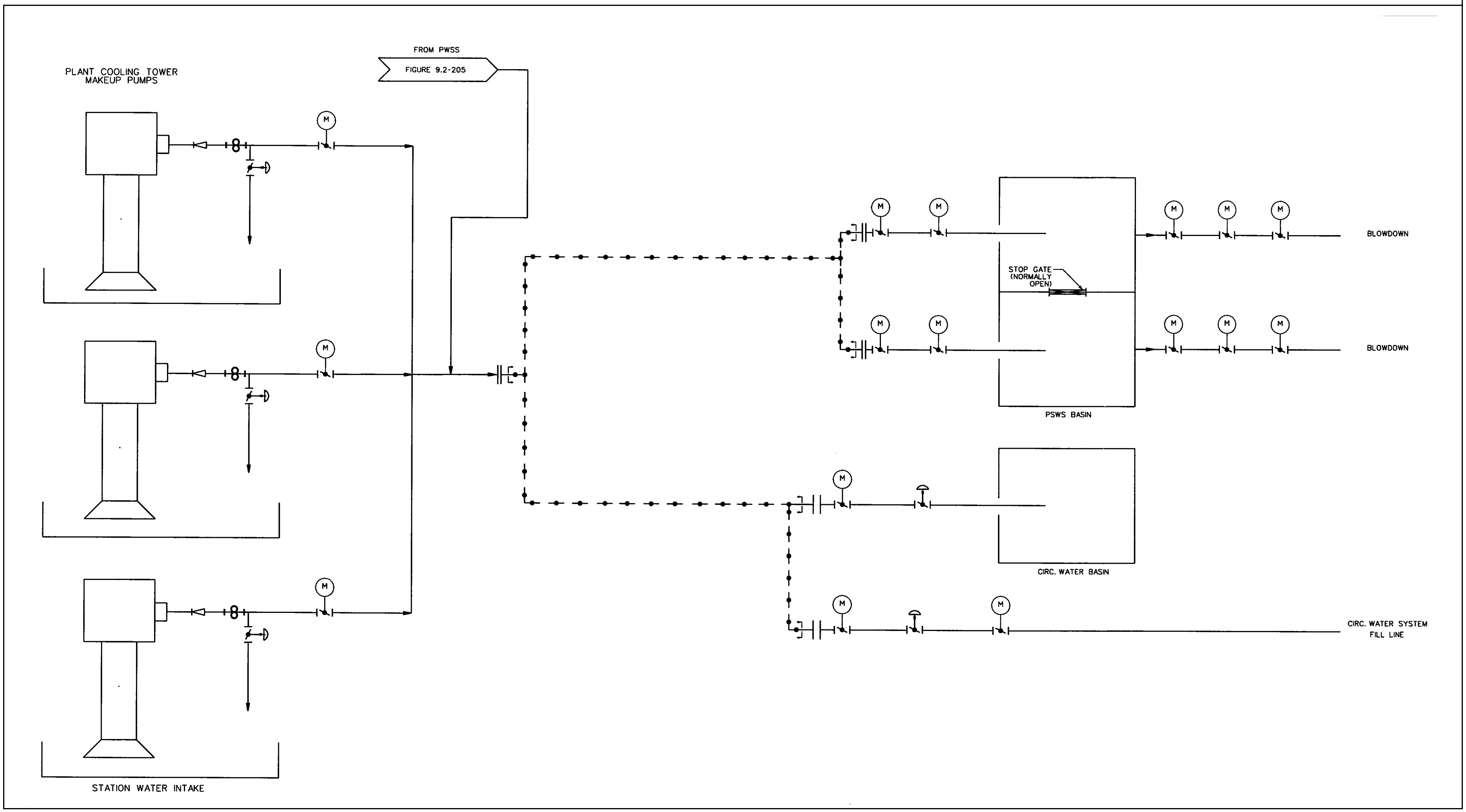
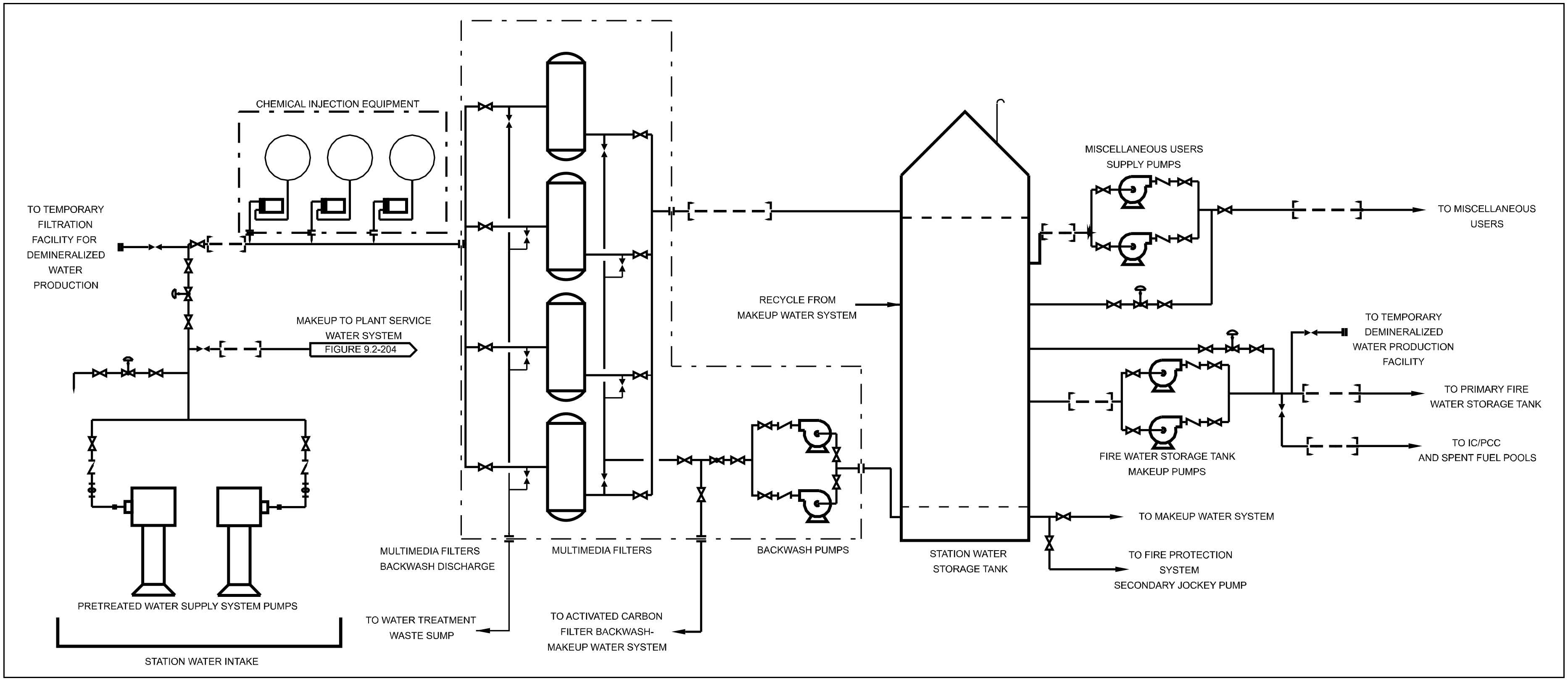


Figure 9.2-205 Station Water System - Pretreated Water Supply System (PWSS)



9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

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

9.3.2 Process Sampling System

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

9.3.2.2 System Description

Add the following at the end of this section.

STD COL 9.3.2-1-A

Post-Accident Sampling Program

The post-accident sampling program consists of the following:

- Emergency Operating Procedures that rely on Emergency Action Levels, defined in the Emergency Plan, are used to classify fuel damage events. These procedures rely on installed post-accident radiation monitoring instrumentation described in DCD Section 7.5 and do not require the capability to obtain and analyze highly radioactive coolant samples although sample analyses may be used for classification as well.
- Plant procedures contain instructions for obtaining highly radioactive grab samples from the following:

  Reactor Coolant - from the RWCU/SDC sample line using the Reactor Building Sample Station. These samples can be analyzed for the parameters indicated in DCD Table 9.3-1. If coolant activity is greater than 1.0 Ci/ml, handling of the samples is delayed to avoid overexposure of personnel.

  Suppression Pool - from FAPCS sample line at the Reactor Building Sample Station. These samples can be analyzed for the parameters indicated in DCD Table 9.3-1. If coolant activity is greater than 1.0 Ci/ml, handling of the samples is delayed to avoid overexposure of personnel.

  Containment Atmosphere - may be taken as described in DCD Section 11.5.3.2.11 and analyzed for fission products.

- DCD Section 7.5.2.2 describes Containment Monitoring System operation in post-LOCA mode for gaseous sampling for O_2 and H_2.
- Effluent radiation monitoring is described in DCD Section 7.5. Field sampling and monitoring capability is maintained in accordance with the Emergency Plan.
- Post accident monitoring is adequate to implement the Emergency Plan without reliance on post accident sampling capability; therefore, the absence of a dedicated Post-Accident Sampling System does not reduce the effectiveness of the Emergency Plan.
- The post-accident sampling program meets the requirements of NUREG-0800, Section 9.3.2 for actions required in lieu of a Post Accident Sampling System.

9.3.2.6 COL Information

9.3.2-1-A Post-Accident Sampling Program

STD COL 9.3.2-1-A

This COL item is addressed in Section 9.3.2.2.

9.3.3 Equipment and Floor Drain System

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

9.3.4 Chemical and Volume Control System

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

9.3.5 Standby Liquid Control System

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

9.3.5.2 System Description

Detailed System Description

Add the following to the end of the fifth paragraph.

STD SUP 9.3.5-1

The above provisions adequately prevent loss of solubility of borated solutions (sodium pentaborate).

9.3.6 Instrument Air System

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

9.3.7 Service Air System

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

9.3.8 High Pressure Nitrogen Supply System

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

9.3.9 Hydrogen Water Chemistry System

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

Replace the first paragraph with the following.

STD COL 9.3.9-1-A The site specific design includes HWCS.

9.3.9.1 Design Basis

Power Generation Design Basis

Replace the first sentence with the following.

STD CDI Hydrogen is added into the feedwater at the suction of the feedwater pumps and oxygen into the offgas system.

9.3.9.2 System Description

Replace this section with the following.

NAPS CDI The HWCS, illustrated in DCD Figure 9.3-5, is composed of hydrogen and oxygen supply systems to inject hydrogen in the feedwater and oxygen in the offgas and several monitoring systems to track the effectiveness of the HWCS. Storage requirements are based on the HWC system usage, ESBWR generator usage and estimated losses.

The hydrogen supply system is integrated with the generator hydrogen supply system (as described in DCD Section 10.2.2.2.8).

NAPS CDI
NAPS COL 9.3.9-2-A

9.3.9.2.1 Hydrogen Storage Facility

The bulk hydrogen storage facility stores liquid hydrogen in an 18,000 gallon vacuum-jacketed pressure vessel. The storage facility is located within a fenced area outside the plant protected area and is open to prevent the accumulation of hydrogen and meets the requirements of DCD References 9.3.9-1 and 9.3.9-2. The hydrogen storage facility consists of a cryogenic tank, cryogenic pumps, atmospheric vaporizers, a compressor, a high-pressure gas storage tubes bank, a hydrogen supply line, pressure regulating valves, an excess flow check valve, and relief valves. The cryogenic tank meets ASME Section VIII, Division 1, requirements for unfired pressure vessels. The pressure regulating valves limit the supply pressure of hydrogen; a relief valve is provided downstream of the regulating valve station to protect the downstream piping in case of regulating valve failure. The excess flow check valve ensures that a large release is limited to the storage facility location. The relief valves provide protection for the storage tank and each isolable liquid hydrogen filled piping section.

The HWCS is implemented with On-line Noble Chem™. Plant personnel conduct the OLNC process while the plant is operating.

The Oxygen Storage Facility is described in Section 9.3.10.2.

9.3.9.4 Inspection and Testing Requirements

Replace this section with the following.

STD CDI The connections for the HWCS are tested and inspected with the feedwater and offgas piping.

Major components of the HWCS are tested and inspected as separate components prior to installation. The system is tested in accordance with vendor requirements after installation to ensure proper performance.

9.3.9.5 Instrumentation and Controls

Replace the first sentence with the following.

STD CDI Instrumentation is provided to control the injection of hydrogen and augment the injection of oxygen.

9.3.9.6 COL Information

9.3.9-1-A Implementation of Hydrogen Water Chemistry

STD COL 9.3.9-1-A This COL item is addressed in Section 9.3.9.

9.3.9-2-A Hydrogen and Oxygen Storage and Supply

NAPS COL 9.3.9-2-A This COL item is addressed in Section 9.3.9.2.1.

9.3.10 **Oxygen Injection System**

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

9.3.10.2 **System Description**

Replace the last paragraph with the following.

NAPS COL 9.3.10-1-A The bulk oxygen storage facility is located outside the plant fenced area. The facility consists of a 34 m^3 (9,000 gal) cryogenic tank, atmospheric vaporizers, an oxygen supply line, a pressure regulating valve, an excess flow check valve, and relief valves. The pressure regulating valve limits the oxygen supply pressure. The excess flow check valve ensures that large releases are limited to the storage facility. The redundant relief valves provide protection for the storage tank and each isolable liquid oxygen filled piping section. The piping carrying gaseous oxygen from the storage facility to the turbine building is routed underground. The storage tank meets ASME Code Section VIII, Division 1, requirements for unfired pressure vessels, and DCD References 9.3.9-1 and 9.3.9-2.

9.3.10.6 **COL Information**

9.3.10-1-A **Oxygen Storage Facility**

NAPS COL 9.3.10-1-A This COL item is addressed in Section 9.3.10.2.

9.3.11 **Zinc Injection System**

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

9.3.11.2 **System Description**

Replace the second paragraph with the following.

STD COL 9.3.11-1-A A Zinc Injection System is not utilized.

9.3.11.4 **Test and Inspections**

Replace the second paragraph with the following.

STD COL 9.3.11-2-A A Zinc Injection System is not utilized.

9.3.11.6 COL Information

9.3.11-1-A Determine Need for Zinc Injection System

STD COL 9.3.11-1-A This COL item is addressed in Section 9.3.11.2.

9.3.11-2-A Provide System Description for Zinc Injection System

STD COL 9.3.11-2-A This COL item is addressed in Section 9.3.11.4.

9.3.12 Auxiliary Boiler System

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

9.4 Heating, Ventilation, and Air Conditioning

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

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

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

9.5.1.1 Design Bases

Codes, Standards, and Regulatory Guidance

Add the following at the end of this section.

NAPS SUP 9.5.1-1 Table 9.5-201 supplements DCD Table 9.5-1 for those portions outside the DCD and operational aspects of the fire detection and suppression systems.

9.5.1.2 System Description

Add the following after the first sentence in the first paragraph.

NAPS COL 9.5.1-4-A Figures 9.5-201, 9.5-202, and 9.5-203 provide simplified diagrams of the site-specific firewater supply piping.

9.5.1.4 Fire Protection Water Supply System

Water Sources

Replace the first paragraph with the following.

NAPS COL 9.5.1-4-A

Water for the Fire Protection System is supplied from a minimum of two sources: i) at least one “primary” source to the suctions of primary fire pumps and corresponding jockey fire pumps and, ii) at least one “secondary” source to suctions of secondary fire pumps and corresponding jockey fire pumps. The primary source is two dedicated, Seismic Category I, firewater storage tanks. Each primary firewater storage tank has sufficient capacity to meet the maximum firewater demand of the system for a period of 120 minutes.

NAPS COL 9.5.1-1-A

The secondary firewater source is Lake Anna. This large body of water has a capacity well in excess of the 2082 m^3 (550,000 gal) required by NFPA 804.

The water from Lake Anna is treated with sodium hypochlorite.

Primary Firewater Source

The Pretreated Water Supply System (PWSS) provides treated and filtered water to the firewater storage tanks. PWSS pumps are located in the Station Water Intake Building. Hypochlorite is added to lake water in the Station Water Intake Building intake bay to preclude biofouling or microbiologically induced corrosion. Strainers are installed at the discharge of the PWSS pumps to preclude large-size foreign materials. The water is also preconditioned to facilitate filtering through multimedia filters before being stored in the station water storage tank and supplied to the firewater storage tanks.

Secondary Firewater Source

The secondary fire pumps are also located in the Station Water Intake Building and draw water from the intake bay. Hypochlorite is added to lake water in the Station Water Intake Building intake bay to preclude biofouling or microbiologically induced corrosion. Hypochlorite can be injected at the discharge of the secondary fire pumps, if required. Strainers are installed at the discharge of secondary firewater pumps to preclude large-size foreign materials. Filtering is not required because of the small amount of total suspended solids in the lake water.

Sampling and monitoring is performed, as required, to ensure an acceptable level of quality of firewater. Periodic system flushes and flow tests are performed to maintain and verify firewater supply system capability.

Water sources that are used for multiple purposes ensure that the required quantity of firewater is dedicated for fire protection use only.

Fire Pumps

Replace the sixth sentence in the first paragraph with the following.

STD COL 9.5.1-2-A

Testing will be performed to demonstrate that the secondary fire protection pump circuit supplies a minimum of 484 m^3/hr (2130 gpm) with sufficient discharge pressure to develop a minimum of 107 psig line pressure at the Turbine Building/yard interface boundary. This cannot be performed until the system is built. This activity will be completed prior to fuel receipt.

9.5.1.5 Firewater Supply Piping, Yard Piping, and Yard Hydrants

Delete the last paragraph and add the following at the end the first paragraph.

NAPS COL 9.5.1-4-A

Figures 9.5-201, 9.5-202, and 9.5-203 provide simplified diagrams of the site-specific firewater supply piping.

9.5.1.10 Fire Barriers

Replace the last paragraph with the following.

STD COL 9.5.1-5-A

Mechanical and electrical penetration seals and electrical raceway fire barrier systems are qualified to the requirements delineated in RG 1.189 by a recognized testing laboratory in accordance with the applicable guidance of NFPA 251 and/or ASTM E-119. Detailed design in this area is not complete. Specific design and certification test results for penetration seal designs and electrical raceway fire barrier systems will be available for review at least six months prior to fuel receipt.

9.5.1.11 Building Ventilation

Replace the last sentence in the third paragraph with the following.

STD COL 9.5.1-6-H

Procedures for manual smoke control will be developed as part of the Fire Protection Program implementation. 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. Other required elements of the Fire Protection Program described in this section are fully operational prior to initial fuel loading per Section 13.4.

9.5.1.12 Safety Evaluation

Replace the first two sentences of the fifth paragraph with the following.

STD COL 9.5.1-7-H

A compliance review of the final as-built design against the assumptions and requirements stated in the FHA will be completed in accordance with the milestones in Section 13.4. Based on this review, the FHA will be updated as necessary.

9.5.1.15 Fire Protection Program

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

STD COL 9.5.1-8-A

The elements of the Fire Protection Program necessary to support receipt and storage of fuel onsite for buildings storing new fuel and adjacent fire areas that could affect the fuel storage area are fully operational prior to receipt for new fuel. Other required elements of the Fire Protection Program described in this section are fully operational prior to initial fuel loading per Section 13.4.

9.5.1.15.1 Fire Protection Program Criteria

Add the following at the end of this section.

NAPS SUP 9.5.1-1

Table 9.5-201 supplements DCD Table 9.5-1.

9.5.1.15.2 [Deleted]

9.5.1.15.3 Fire Protection Program Staffing Requirements

Replace this section with the following.

NAPS COL 13.1-1-A

Fire protection staffing and organization of the fire brigade are described in Section 13.1.

9.5.1.15.4 Onsite Fire Operations Training

Replace the first paragraph with the following.

NAPS COL 9.5.1-10-H

Implementation of the fire brigade will be in accordance with the milestones in Section 13.4 for the Fire Protection Program.

9.5.1.15.6 Control of Combustible Materials, Hazardous Materials and Ignition Sources

Add the following at the end of this section.

STD SUP 9.5.1-3

- In rooms adjacent to the main control room and in computer rooms that are not part of the control room complex:
  - Transient combustible materials are not left unattended during lunch breaks, shift changes, or other similar periods unless stored in approved containers.
  - Electrical appliances and other potential ignition sources are controlled.
- Prohibit the storage of transient combustibles below the raised floor in the main control complex.
- Prohibit the storage of hazardous chemicals in areas that contain or expose equipment important to safety.

9.5.1.15.9 Quality Assurance

Replace this section with the following.

STD COL 9.5.1-11-A

Quality assurance controls are applied to the activities involved in the design, procurement, installation, and testing and the administrative controls of fire protection systems, in accordance with the measures outlined in Chapter 17.

For the operational fire protection program, the Quality Assurance Program implements the requirements of RG 1.189 through site-specific administrative controls procedures. The procedures will be developed six months prior to fuel receipt and will be fully implemented prior to fuel receipt.

9.5.1.16 COL Information

9.5.1-1-A Secondary Firewater Storage Source

NAPS COL 9.5.1-1-A This COL item is addressed in Section 9.5.1.4 and DCD Table 9.5-2.

9.5.1-2-A Secondary Firewater Capacity

NAPS COL 9.5.1-2-A This COL item is addressed in Section 9.5.1.4.

9.5.1-4-A Piping and Instrument Diagrams

NAPS COL 9.5.1-4-A This COL item is addressed in Sections 9.5.1.2, 9.5.1.4, 9.5.1.5, and Figures 9.5-201, 9.5-202, and 9.5-203.

9.5.1-5-A Fire Barriers

STD COL 9.5.1-5-A This COL item is addressed in Section 9.5.1.10.

9.5.1-6-H Smoke Control

STD COL 9.5.1-6-H This COL item is addressed in Section 9.5.1.11.

9.5.1-7-H FHA Compliance Review

STD COL 9.5.1-7-H This COL item is addressed in Section 9.5.1.12.

9.5.1-8-A FP Program Description

STD COL 9.5.1-8-A This COL item is addressed in Section 9.5.1.15.

9.5.1-9-A [Deleted]

9.5.1-10-H Fire Brigade

NAPS COL 9.5.1-10-H This COL item is addressed in Sections 9.5.1.15.4 and 13.1.2.1.5.

9.5.1-11-A Quality Assurance

STD COL 9.5.1-11-A This COL item is addressed in Section 9.5.1.15.9.

DCD Table 9.5-2

NAPS COL 9.5.1-1-A Delete the “*” and “**” footnotes.

9.5.2 Communications System

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

9.5.2.2 System Description

Emergency Communication Systems

Replace the parenthetical "(COL 9.5.2.5-1-A)" in the first bullet with the following.

NAPS COL 9.5.2.5-1-A The North Anna Emergency Notification System (ENS) is provided in the plant Emergency Plan. The ENS phone lines are routed directly to the local telephone company central office via fiber-optic phone lines through a telephone utility switch that is located on site in the telephone equipment building. The normal power for this device is non-safety related station power. The telephone system will lose its normal power supply during a loss of offsite power; however, the phone system is battery backed for a period of approximately eight hours. This design ensures that the ENS located at the site is fully operable from the site in the event of a loss of offsite power at the site and is in compliance with the requirements of NRC Bulletin 80-15 for the ENS. Automatic Ringdown Circuits (ARD) (described in the plant Emergency Plan) connect the plant to the local and state emergency offices, and are also normally powered from the non-safety related station power and backed with approximately eight hours of battery backup power. In addition to the connections to the local telephone company, a separate Company-owned and maintained fiber-optic network exists which provides communication between the station, the system operations center, and the NRC. This Company network is also capable of external long distant and local telephone calls.

Replace the parenthetical "(COL 9.5.2.5-3-A)" in the second bullet with the following.

NAPS COL 9.5.2.5-3-A The health physics network is described in the Emergency Plan.

Replace the parenthetical “(COL 9.5.2.5-4-A)” in the third bullet with the following.

NAPS COL 9.5.2.5-4-A Communication from the Control Room, TSC, and EOF to NRC headquarters including establishment of Emergency Response Data Systems (ERDS) is described in the Emergency Plan.

Replace the parenthetical “(COL 9.5.2.5-3-A)” in the fourth bullet with the following.

NAPS COL 9.5.2.5-3-A The crisis management radio system is part of the plant radio system described in DCD Section 9.5.2.2.

Replace the parenthetical “(COL 9.5.2.5-5-A)” in the fifth bullet with the following.

NAPS COL 9.5.2.5-5-A The fire brigade radio system is part of the plant radio system described in DCD Section 9.5.2.2.

Replace the last bullet with the following.

NAPS COL 9.5.2.5-2-A

- Transmission System Operator Communications Link: Voice communications with the grid operator are provided via a Company-owned and -maintained fiber optic transmission system that allows telephone communications with the entire Corporate System. Access to this mode of transmission is made via the plant telephone system. A dedicated handset is provided between the Control Room and the power system operator.

Add the following after the last bullet.

NAPS COL 9.5.2.5-3-A

- Insta-Phone System - The primary method for notification of State and local authorities is the Insta-phone, which is accessible from the Control Room, TSC, and EOF. The Insta-phone is described in the Emergency Plan.

9.5.2.5 COL Information

9.5.2.5-1-A Emergency Notification System

NAPS COL 9.5.2.5-1-A This COL item is addressed in Section 9.5.2.2.

9.5.2.5-2-A **Grid Transmission Operator**

NAPS COL 9.5.2.5-2-A

This COL item is addressed in Section 9.5.2.2 and Emergency Plan Section II.F.1.

9.5.2.5-3-A **Offsite Interfaces (1)**

NAPS COL 9.5.2.5-3-A

This COL item is addressed in Section 9.5.2.2 and Emergency Plan Sections II.E.1 and II.F.1.

9.5.2.5-4-A **Offsite Interfaces (2)**

NAPS COL 9.5.2.5-4-A

This COL item is addressed in Section 9.5.2.2 and Emergency Plan Sections II.E.1 and II.F.1.

9.5.2.5-5-A **Fire Brigade Radio System**

NAPS COL 9.5.2.5-5-A

This COL item is addressed in Section 9.5.2.2.

9.5.3 Lighting System

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

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

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

9.5.4.2 System Description

Detailed System Description

Standby Diesel Generators

Replace the third to last sentence in the first paragraph with the following.

STD COL 9.5.4-1-A

Procedures require that the quantity of diesel fuel oil in the standby diesel generator (SDG) fuel oil storage tanks is monitored on a periodic basis. The diesel fuel oil usage is tracked against planned deliveries. Regular transport replenishes the diesel fuel oil inventory during periods of high demand and ensures continued supply in the event of adverse weather conditions. These procedures ensure sufficient diesel fuel oil inventory is available on site so that the SDGs can operate continually for seven days with each operating at its calculated design load, with appropriate design margins. The procedures will be developed in accordance with the milestone and processes described in Section 13.5.

Replace the third paragraph with the following.

NAPS COL 9.5.4-2-A The only underground component of the SDGs fuel oil storage and transfer system is carbon steel piping. A corrosion protection system is provided for external surfaces of buried piping systems. The buried sections of the piping are provided with waterproof protective coating and an impressed current type cathodic protection to control external corrosion.

STD COL 9.5.4-1-A Delete the parenthetical "(COL 9.5.4-1-A)" at the end of the last paragraph.

Ancillary Diesel Generators

Replace the third to last sentence in the first paragraph with the following.

STD COL 9.5.4-1-A Procedures require that the quantity of diesel fuel in the ancillary diesel generator (ADG) fuel oil storage tanks is monitored on a periodic basis. The diesel fuel oil usage is tracked against planned deliveries. Regular transport replenishes the fuel oil inventory during periods of high demand and ensures continued supply in the event of adverse weather conditions. These procedures ensure sufficient diesel fuel oil inventory is available on site so that the ADGs can operate continually for seven days with each operating at its calculated design load, with appropriate design margins. The procedures will be developed in accordance with the milestone and processes described in Section 13.5.

Replace the third paragraph with the following.

NAPS COL 9.5.4-2-A The only underground component of the ADGs fuel oil storage and transfer system is carbon steel piping. A corrosion protection system is provided for external surfaces of buried piping systems. The buried sections of the piping are provided with waterproof protective coating and an impressed current type cathodic protection to control external corrosion.

System Operation

Standby Diesel Generators

STD COL 9.5.4-1-A Delete the parenthetical "(COL 9.5.4-1-A)" at the end of the paragraph.

Ancillary Diesel Generators

STD COL 9.5.4-1-A Delete the parenthetical "(COL 9.5.4-1-A)" at the end of the paragraph.

9.5.4.6 COL Information

9.5.4-1-A Fuel Oil Capacity

STD COL 9.5.4-1-A This COL item is addressed in Section 9.5.4.2.

9.5.4-2-A Protection of Underground Piping

NAPS COL 9.5.4-2-A This COL item is addressed in Section 9.5.4.2.

9.5.5 Diesel Generator Jacket Cooling Water System

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

9.5.6 Diesel Generator Starting Air System

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

9.5.7 Diesel Generator Lubrication System

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

9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

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

NAPS SUP 9.5.1-1
NAPS SUP 9A-01

Table 9.5-201 Codes and Standards

American Society of Mechanical Engineers (ASME)

Boiler and Pressure Vessel Code	Section IX, Qualification Standard for Welding and Brazing Procedures, Welder, Brazers and Welding and Brazing Operators

Applicable Building Codes

Virginia Uniform Statewide Building Code	Virginia Uniform Statewide Building Code, Part I (Virginia Construction Code) As defined in the Virginia Uniform Statewide Building Code edition of record.

National Fire Protection Association (NFPA)

NFPA 1	Uniform Fire Code
NFPA 25	Recommended Practices for Inspection, Testing, and Maintenance of Standpipes and Hose Systems
NFPA 55	Standard for Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks
NFPA 259	Standard Test Method for Potential Heat of Building Materials
NFPA 703	Standard for Fire-Retardant Treated Wood and Fire Retardant Coatings for Building Materials
NFPA 750	Standard for Water Mist Fire Protection Systems
NFPA 1144	Standard for Reducing Structure Ignition Hazards from Wildland Fire
NFPA 1410	Standard on Training for Initial Emergency Scene Operations
NFPA 1620	Recommended Practice for Pre-Incident Planning
NFPA 2001	Standard for Clean Agent Fire Extinguishing

Environmental Protection Agency (EPA)

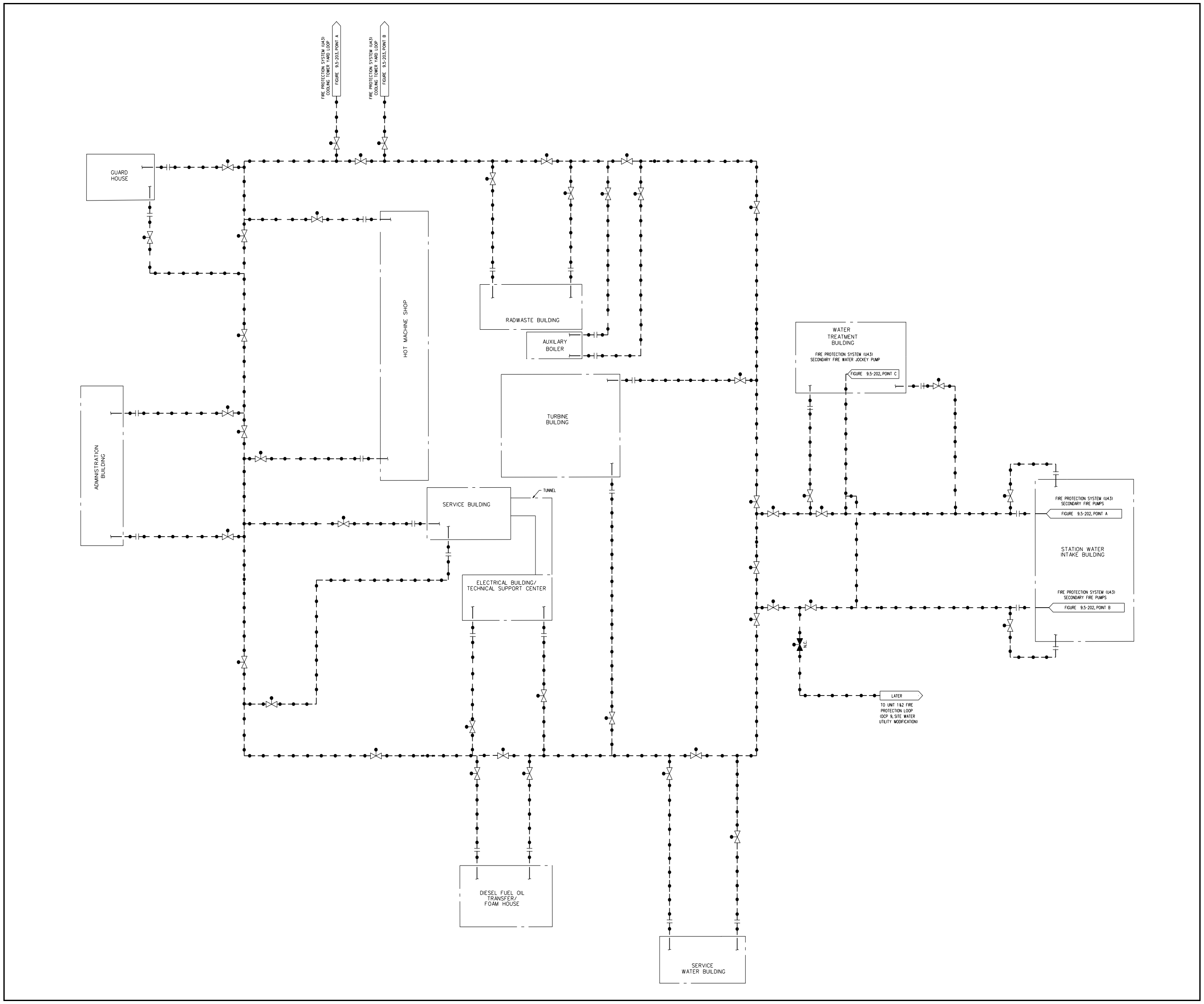
Environmental Protection Agency (EPA)	EPA Standards of Performance for Stationary Compression Ignition Internal Combustion Engines; Final Rule (40 CFR Parts 60, 85 et al.)

Listing/Approval Agencies

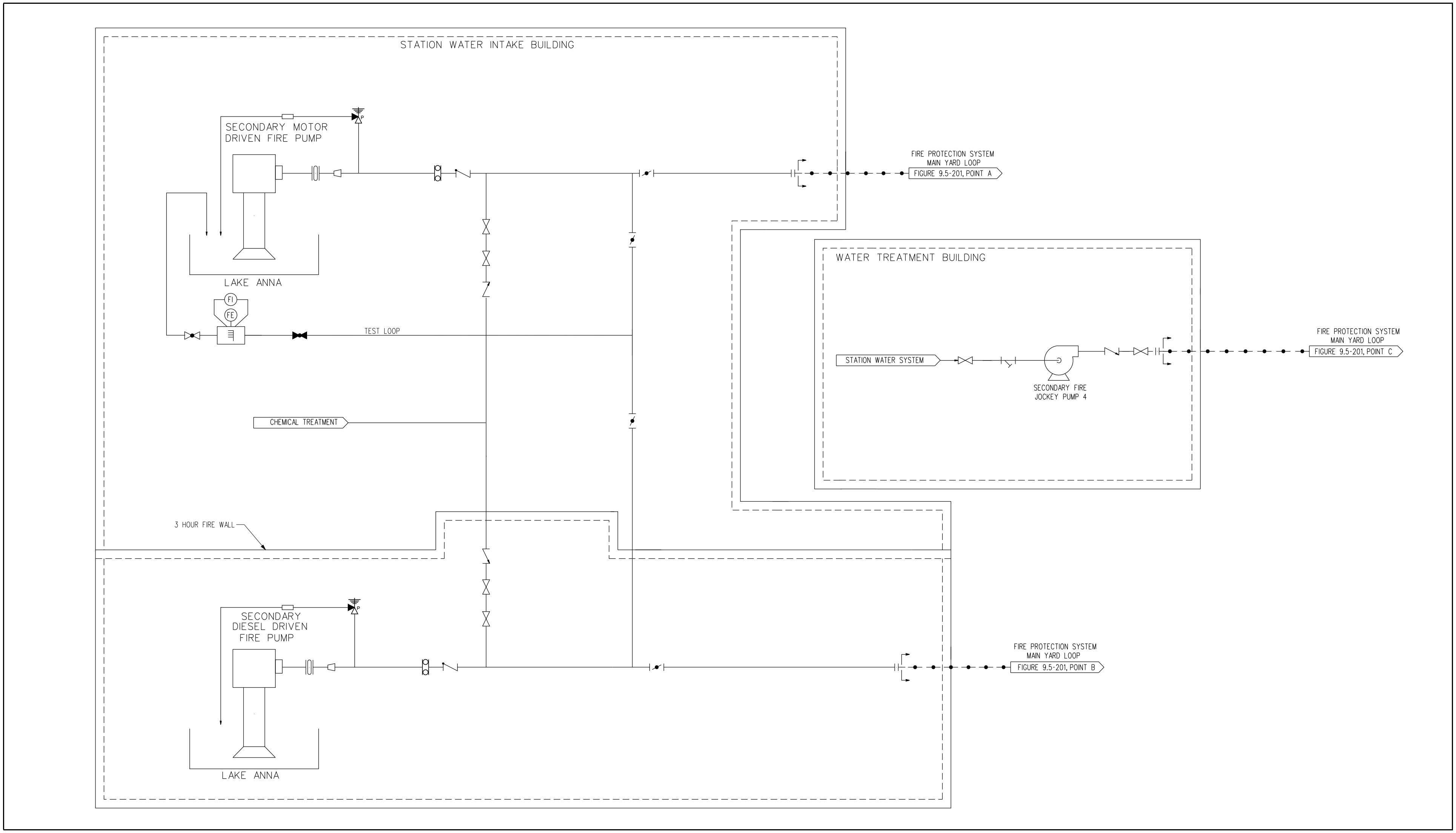
Nuclear Electric Insurance Limited (NEIL)

NAPS COL 9.5.1-4A

Figure 9.5-201 Fire Protection System; Main Yard Loop

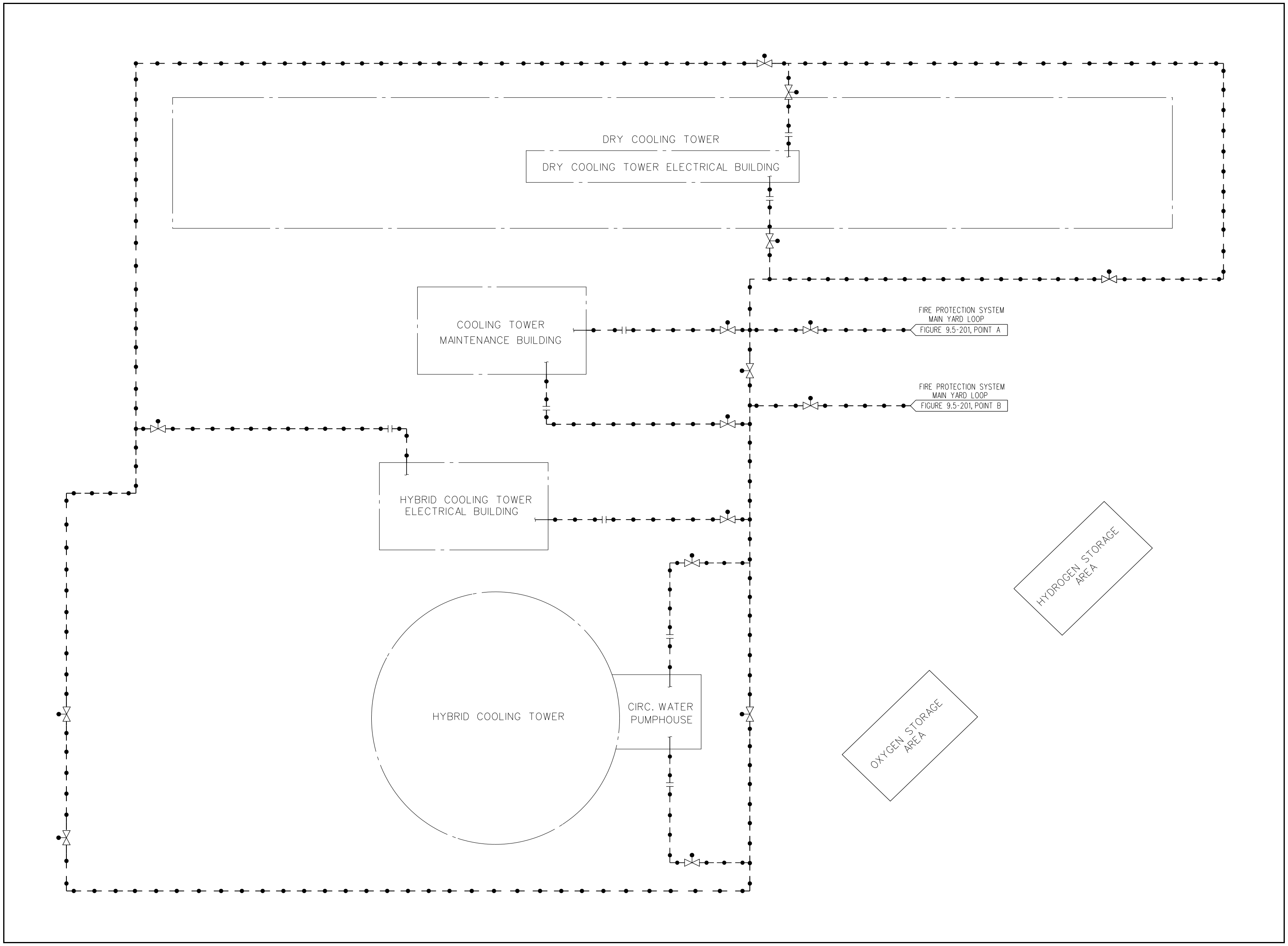


NAPS COL 9.5.1-4-A **Figure 9.5-202 Fire Protection System Secondary Fire Pumps**



NAPS COL 9.5.1-4-A

Figure 9.5-203 Fire Protection System; Cooling Tower Yard Loop



Appendix 9A Fire Hazards Analysis

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

Contents

NAPS CDI

Replace 9A.4.9 Service Water/Water Treatment Building with 9A.4.9 Service Water Building.

Replace 9A.5.9 Service Water/Water Treatment Building with 9A.5.9 Service Water Building.

Add 9A.5.12, Water Treatment Building

9A.1 Introduction

NAPS CDI

In the first sentence of the first paragraph, replace "Service Water/Water Treatment Building" with "Service Water Building, Water Treatment Building,"

and

Replace "Pump House" with "Circulating Water Pump House, Station Water Intake Building."

NAPS CDI

In the first sentence of the first paragraph, delete "Cold Machine Shop, Warehouse."

9A.2.1 Codes and Standards

Add the following second paragraph.

NAPS SUP 9A-01

The codes and standards that are applicable to the design of the site-specific portions of the yard are listed in Table 9.5-201. Table 1.9-204 identifies the relevant editions for each applicable code and standard. These codes and standards also apply to the operational aspects of the fire detection and suppression systems.

9A.3.1 Review Data

NAPS CDI

In the second paragraph, first sentence replace “Pump House” with “Circulating Water Pump House, Station Water Intake Building.”

and

Replace “Service Water/Water Treatment Building” with “Service Water Building, Water Treatment Building.”

NAPS CDI

In the first sentence of the second paragraph, delete “Cold Machine Shop, Warehouse.”

9A.4.7 Yard

Replace the first paragraph with the following.

STD COL 9A.7-1-A

The Yard includes all portions of the plant site external to the Reactor Building, Fuel Building, Control Building, Turbine Building, Radwaste Building, and Electrical Building. The fire zone drawings for the site-specific portions of the yard are provided in Figures 9A.2-201 through 9A.2-206.

Replace the second paragraph with the following.

NAPS COL 9A.7-2-A

A detailed fire hazards analysis of the yard area that is outside the scope of the certified design can not be completed until cable routing is performed during final design. This information will be provided six months prior to fuel load. The FSAR will be revised to include this information, as appropriate, as part of a subsequent FSAR update.

NAPS CDI

In the first sentence of the third paragraph, delete “Cold Machine Shop, Warehouse.”

Delete the eighth paragraph.

9A.4.9 Service Water/Water Treatment Building

NAPS CDI

Replace the title with “Service Water Building.”

In the first sentence of the first paragraph, replace “Service Water/Water Treatment Building (SF/WT)” with “Service Water Building.”

Replace “SF/WT” with “Service Water Building” in the first, second, and third paragraphs.

9A.5.7 Yard

Replace the last two sentences with the following.

NAPS COL 9A.7-2-A

A detailed fire hazards analysis of the yard area that is outside the scope of the certified design can not be completed until cable routing is performed during final design. This information will be provided six months prior to fuel load. The FSAR will be revised to include this information, as appropriate, as part of a subsequent FSAR update.

9A.5.8 Service Building

Replace the last two sentences with the following.

NAPS CDI
NAPS COL 9A.7-2-A

A detailed fire hazards analysis of the yard area that is outside the scope of the certified design, which includes the Service Building, can not be completed until cable routing is performed during final design. This information will be provided six months prior to fuel load. The FSAR will be revised to include this information, as appropriate, as part of a subsequent FSAR update.

9A.5.9 Service Water/Water Treatment Building

NAPS CDI

Replace the title with “Service Water Building.”

Replace this section with the following.

NAPS COL 9A.7-2-A

The Service Water Building is protected in accordance with applicable codes. The Service Water Building contains service water equipment which has RTNSS functions. A detailed fire hazards analysis of the yard area that is outside the scope of the certified design, which includes the Service Water Building, can not be completed until cable routing is performed during final design. This information will be provided six months prior to fuel load. The FSAR will be revised to include this information, as appropriate, as part of a subsequent FSAR update.

NAPS CDI
NAPS COL 9A.7-2-A

9A.5.12 Water Treatment Building

The Water Treatment Building is protected in accordance with applicable NFPA Codes. The Water Treatment Building is site specific.

A detailed fire hazards analysis of the yard area that is outside the scope of the certified design, which includes the Water Treatment Building, can not be completed until cable routing is performed during final design. This information will be provided six months prior to fuel load. The FSAR will be revised to include this information, as appropriate, as part of a subsequent FSAR update.

9A.7 COL Information

9A.7-1-A Yard Fire Zone Drawings

NAPS COL 9A.7-1-A This COL item is addressed in Section 9A.4.7.

9A.7-2-A FHA for Site-Specific Areas

NAPS COL 9A.7-2-A This COL item is addressed in Sections 9A.4.7, 9A.5.7, 9A.5.8, 9A.5.9, and 9A.5.12.

Table 9A.5-7 Revisions

NAPS COL 9A.7-2-A Delete Fire Area F4202.

Replace Fire Area F5159 with F5159R.

Replace Fire Area F5169 with F5169R.

Delete Fire Area F7100.

Add Fire Areas F7151, F7152, F7153, F7154, F7161, F7162, F7163, F7164, F7174, F7165, and F7155.

Add Fire Area F7180.

Add Fire Area F7188.

Delete Fire Area F7200.

Delete Fire Area F7300.

Add Fire area F7301, F7302, F7303, and F7304.

Add Fire Area F7305.

Delete Fire Area F7400.

Delete Fire Area F7500

Replace Fire Area F7700 with F7700R.

Replace Fire Area F7900 with F7900R.

Add Fire Areas F8101, F8102, and F8103.

Add Fire Areas F8104, F8105, F8106 and F8108.

Add Fire Areas F8181, F8282, F8183, F8184, F8185, F8186, F8187, F8188, and F8283.

Add Fire Areas F8200, F8201, F8107, F8109 and F8189.

Delete all fire areas designated as “site specific.”

NAPS COL 9A.7-2-A

Table 9A.5-7R Yard

Fire Area	F5159R
Description	Fuel Oil Storage Tank A
Building	Diesel Tanks
Fire Zone Dwg	9A.2-33R
Applicable Codes	IBC; Reg Guide 1.189; NFPA 11, 16, 24, 30, 72, 804
Building code occupancy classification	U
Electrical classification	N/A
Safety-related divisional equipment or cables	N/A
Non-safety-related redundant trains or equipment or cables	A
Surrounded by fire barriers rated at	N/A
Except	

Consisting of the following rooms:

Elevation	Room #	Potential Combustibles	Fire Detection		Fire Suppression	
			Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	215,400 gal Class II Fuel Oil	Spot Heat Inside Tank	Manual Pulls	Foam Injection - Manual Release	Foam Hose Stations

Anticipated combustible load, MJ/m^2	> 700
Non-sprinkled combustible load limit, MJ/m^2	700

Assuming operation of fire suppression systems, effect of fire upon:

Plant operation	None
Radiological release	None, no radiological materials present
Life safety	N/A
Manual firefighting	Access all around
Property loss	Moderate

Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:

Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.

NAPS COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F5169R					
Description	Fuel Oil Storage Tank B					
Building	Diesel Tanks					
Fire Zone Dwg	9A.2-33R					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 11, 16, 24, 30, 72, 804					
Building code occupancy classification					U	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					B	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	215,400 gal Class II Fuel Oil	Spot Heat Inside Tank	Manual Pulls	Foam Injection - Manual Release	Foam Hose Stations
Anticipated combustible load, MJ/m^2					> 700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	N/A					
Manual firefighting	Access all around					
Property loss	Moderate					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

NAPS COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F7151					
Description	Pump Room Train A					
Building	Service Water Building					
Fire Zone Dwg	9A.2-204					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					A	
Surrounded by fire barriers rated at					3-hour	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation	Manual Pulls (at EXITs)	None	Fire Extinguishers	Yard Hydrants
Anticipated combustible load, MJ/m^2					LATER	
Non-sprinkled combustible load limit, MJ/m^2					LATER	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

NAPS
COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area		F7152				
Description		Electrical Room Train A				
Building		Service Water Building				
Fire Zone Dwg		9A.2-204				
Applicable Codes		IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804				
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					A	
Surrounded by fire barriers rated at					3-hour	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation	Smoke	Manual Pulls (at EXITs)	Preaction Sprinkler LATER L/min per m^2	CO_2 Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2					> 700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation		To be determined during detailed design				
Radiological release		None, no radiological materials present				
Life safety		To be determined during detailed design				
Manual firefighting		To be determined during detailed design				
Property loss		To be determined during detailed design				
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

NAPS COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area		F7153				
Description		Cooling Tower Train A				
Building		Service Water Building				
Fire Zone Dwg		9A.2-204				
Applicable Codes		IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804				
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					A	
Surrounded by fire barriers rated at					3-hour	
Except						
Consisting of the following rooms:						
Elevation	Room #	Potential Combustibles	Fire Detection		Fire Suppression	
			Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation Fill Material	Manual Pulls (at EXITs)	None	Fire Extinguishers	Yard Hydrants
Anticipated combustible load, MJ/m^2					LATER	
Non-sprinkled combustible load limit, MJ/m^2					LATER	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation		To be determined during detailed design				
Radiological release		None, no radiological materials present				
Life safety		To be determined during detailed design				
Manual firefighting		To be determined during detailed design				
Property loss		To be determined during detailed design				
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

NAPS
COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F7154					
Description	Transfer Pump Room A					
Building	Diesel Fuel Oil Transfer/Foam House					
Fire Zone Dwg	9A.2-202					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					A	
Surrounded by fire barriers rated at					3-hour	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	No. 2 Diesel Fuel Oil Cable Insulation Electrical Equipment	Manual Pulls (at EXITS)	None	Foam Hose Racks	Fire Extinguishers
Anticipated combustible load, MJ/m^2					LATER	
Non-sprinkled combustible load limit, MJ/m^2					LATER	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

NAPS COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F7155					
Description	Electrical Room A					
Building	Diesel Fuel Oil Transfer/Foam House					
Fire Zone Dwg	9A.2-202					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					A	
Surrounded by fire barriers rated at					3-hour	
Except					Exterior Walls (non-rated)	
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Electrical Equipment	Area Wide Ionization	Manual Pulls	CO_2 Fire Extinguishers	Hose Racks
Anticipated combustible load, MJ/m^2					< 1400	
Non-sprinkled combustible load limit, MJ/m^2					1400	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

NAPS
COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F7161					
Description	Pump Room Train B					
Building	Service Water Building					
Fire Zone Dwg	9A.2-204					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					B	
Surrounded by fire barriers rated at					3-hour	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation	Manual Pulls (at EXITS)	None	Fire Extinguishers	Yard Hydrants
Anticipated combustible load, MJ/m^2					LATER	
Non-sprinkled combustible load limit, MJ/m^2					LATER	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

NAPS COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F7162					
Description	Electrical Room Train B					
Building	Service Water Building					
Fire Zone Dwg	9A.2-204					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					B	
Surrounded by fire barriers rated at					3-hour	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation	Manual Pulls (at EXITS)	None	Fire Extinguishers	Yard Hydrants
Anticipated combustible load, MJ/m^2					<1400	
Non-sprinkled combustible load limit, MJ/m^2					1400	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

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Table 9A.5-7R Yard (continued)

Fire Area		F7163				
Description		Cooling Tower Train B				
Building		Service Water Building				
Fire Zone Dwg		9A.2-204				
Applicable Codes		IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804				
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					B	
Surrounded by fire barriers rated at					3-hour	
Except						
Consisting of the following rooms:						
		Potential	Fire Detection		Fire Suppression	
Elevation	Room #	Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation Fill Material	Manual Pulls (at EXITS)	None	Fire Extinguishers	Yard Hydrants
Anticipated combustible load, MJ/m^2					LATER	
Non-sprinkled combustible load limit, MJ/m^2					LATER	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation		To be determined during detailed design				
Radiological release		None, no radiological materials present				
Life safety		To be determined during detailed design				
Manual firefighting		To be determined during detailed design				
Property loss		To be determined during detailed design				
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

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Table 9A.5-7R Yard (continued)

Fire Area	F7164					
Description	Transfer Pump Room B					
Building	Diesel Fuel Oil Transfer/Foam House					
Fire Zone Dwg	9A.2-202					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					B	
Surrounded by fire barriers rated at					3-hour	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	No. 2 Diesel Fuel Oil Cable Insulation Electrical Equipment	Manual Pulls (at EXITS)	None	Foam Hose Racks	Fire Extinguishers
Anticipated combustible load, MJ/m²					LATER	
Non-sprinkled combustible load limit, MJ/m²					LATER	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

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Table 9A.5-7R Yard (continued)

Fire Area	F7165					
Description	Electrical Room B					
Building	Diesel Fuel Oil Transfer/Foam House					
Fire Zone Dwg	9A.2-202					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					B	
Surrounded by fire barriers rated at					3-hour	
Except					Exterior Walls (non-rated)	
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Electrical Equipment	Area Wide Ionization	Manual Pulls	CO_2 Fire Extinguishers	Hose Racks
Anticipated combustible load, MJ/m^2					< 1400	
Non-sprinkled combustible load limit, MJ/m^2					1400	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and/or cables in this fire area will not affect any safety-related, safe shutdown, or RTNSS divisional equipment and/or cables outside of this fire area.						

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Table 9A.5-7R Yard (continued)

Fire Area	F7174					
Description	Foam House					
Building	Diesel Fuel Oil Transfer/Foam House					
Fire Zone Dwg	9A.2-202					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 10, 13, 72, 75, 90A, 101, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					3-hour	
Except						
Consisting of the following rooms:						
		Potential	Fire Detection		Fire Suppression	
Elevation	Room #	Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	LATER	Manual Pulls (at EXITS)	None	Fire Extinguisher	Foam Hose Rack
Anticipated combustible load, MJ/m^2					< 700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area	F7301					
Description	General Area					
Building	Water Treatment Building					
Fire Zone Dwg	9A.2-201					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					H-4	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation Plastic Filter Membranes Corrosive/ Toxic Chemicals	Manual Pulls (at EXITS)	None	Wet-Pipe Sprinkler LATER L/min per m^2	Hose Racks Portable Extinguishers
Anticipated combustible load, MJ/m^2					>700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None, but may affect makeup water chemistry					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment, but could affect nonsafety-related equipment including equipment which could be used for make-up to IC/PCCS pools and spent fuel pool if 7 days post accident; all safety divisions and both on-site and off-site power supplies are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area	F7302					
Description	Electrical Room					
Building	Water Treatment Building					
Fire Zone Dwg	9A.2-201					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					1 hour per IBC table 302.3.2	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation Electrical Equipment	Smoke	Manual Pulls (at EXITs)	Pre-Action Sprinkler LATER L/min per m^2	Hose Racks Portable Extinguishers
Anticipated combustible load, MJ/m^2					>1400	
Non-sprinkled combustible load limit, MJ/m^2					1400	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None, but may affect makeup water chemistry					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment, but could affect nonsafety-related equipment including equipment which could be used for make-up to IC/PCCS pools and spent fuel pool 7 days post accident; all safety divisions and both on-site and off-site power supplies are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area	F7303
Description	Control Room
Building	Water Treatment Building
Fire Zone Dwg	9A.2-201
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804
Building code occupancy classification	F-1
Electrical classification	N/A
Safety-related divisional equipment or cables	N/A
Non-safety-related redundant trains or equipment or cables	N/A
Surrounded by fire barriers rated at	1 hour per IBC table 302.3.2
Except	Outside walls

Consisting of the following rooms:

Elevation	Room #	Potential Combustibles	Fire Detection		Fire Suppression	
			Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation Electrical Equipment	Smoke	Manual Pulls (at EXITs)	Pre-Action Sprinkler LATER L/min per m^2	Hose Racks Portable Extinguishers

Anticipated combustible load, MJ/m^2	>700
Non-sprinkled combustible load limit, MJ/m^2	700

Assuming operation of fire suppression systems, effect of fire upon:

Plant operation	None, but may affect makeup water chemistry
Radiological release	None, no radiological materials present
Life safety	To be determined during detailed design
Manual firefighting	To be determined during detailed design
Property loss	To be determined during detailed design

Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:

Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment, but could affect nonsafety-related equipment including equipment which could be used for make-up to IC/PCCS pools and spent fuel pool 7 days post accident; all safety divisions and both on-site and off-site power supplies are unaffected by fire and are operable.

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Table 9A.5-7R Yard (continued)

Fire Area	F7304					
Description	Lab					
Building	Water Treatment Building					
Fire Zone Dwg	9A.2-201					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					1 hour per IBC table 302.3.2	
Except					Outside walls	
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Electrical Equipment Cable Insulation	Smoke	Manual Pulls (at EXITs)	Pre-Action Sprinkler LATER L/min per m^2	Hose Racks Portable Extinguishers
Anticipated combustible load, MJ/m^2					>700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None, but may affect makeup water chemistry					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment, but could affect nonsafety-related equipment including equipment which could be used for make-up to IC/PCCS pools and spent fuel pool 7 days post accident; all safety divisions and both on-site and off-site power supplies are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area		F7305				
Description		Circulating Water Pump House				
Building		Circulating Water Pump House				
Fire Zone Dwg		9A.2-33R				
Applicable Codes		IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804				
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except					N/A	
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	LATER	Manual Pulls (at EXITs)	None	LATER	LATER
Anticipated combustible load, MJ/m^2					< 700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation		To be determined during detailed design				
Radiological release		None, no radiological materials present				
Life safety		To be determined during detailed design				
Manual firefighting		To be determined during detailed design				
Property loss		To be determined during detailed design				
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

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Table 9A.5-7R Yard (continued)

Fire Area	F7180					
Description	Guard House					
Building	Guard House					
Fire Zone Dwg	9A.2-33R					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 10, 24, 72, 90A, 101, 804					
Building code occupancy classification	B					
Electrical classification	N/A					
Safety-related divisional equipment or cables	N/A					
Non-safety-related redundant trains or equipment or cables	N/A					
Surrounded by fire barriers rated at	N/A					
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation	Smoke	Manual Pulls (at EXITs)	Wet-Pipe Sprinkler	Fire Extinguishers
Anticipated combustible load, MJ/m^2	> 700					
Non-sprinkled combustible load limit, MJ/m^2	700					
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

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Table 9A.5-7R Yard (continued)

Fire Area	F7188					
Description	Chemical Storage Area					
Building	Service Water Building					
Fire Zone Dwg	9A.2-204					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification	F-1					
Electrical classification	N/A					
Safety-related divisional equipment or cables	N/A					
Non-safety-related redundant trains or equipment or cables	N/A					
Surrounded by fire barriers rated at	1-hour					
Except	Exterior Walls (non-rated)					
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Corrosive/ Toxic Chemicals	Manual Pulls (at EXITs)	None	Wet-Pipe Sprinkler LATER L/min per m^2	Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2	>700					
Non-sprinkled combustible load limit, MJ/m^2	700					
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area	F7700R					
Description	Service Building					
Building	Service Building					
Fire Zone Dwg	9A.2-33R					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 10, 13, 72, 90A, 101, 804; 28 CFR 36					
Building code occupancy classification					B	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					3-hour	
Except					South, East, North Walls (non-rated)	
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Class A Combustibles Cable Insulation	Smoke	Manual Pulls (at EXITs)	Wet-Pipe Sprinkler	ABC Fire Extinguisher
Anticipated combustible load, MJ/m^2					> 700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

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Table 9A.5-7R Yard (continued)

Fire Area	F7900R					
Description	Administration Building					
Building	Administration Building					
Fire Zone Dwg	9A.2-33R					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 10, 13, 72, 90A, 101, 804					
Building code occupancy classification					B	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Class A Combustibles Cable Insulation	Suppression Flowswitch	Manual Pulls (at EXITs)	Wet-Pipe Sprinkler	Fire Extinguishers Hose Racks
Anticipated combustible load, MJ/m^2					> 700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

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Table 9A.5-7R Yard (continued)

Fire Area	F8101					
Description	Motor Driven Fire Pump (Intake Area)					
Building	Station Water Intake Building					
Fire Zone Dwg	9A.2-203					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 10, 13, 20, 24, 30, 37, 72, 101, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					3-hour	
Except					Exterior Walls (non-rated)	
Consisting of the following rooms:						
Elevation	Room #	Potential Combustibles	Fire Detection		Fire Suppression	
			Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation	Suppression Flowswitch	Manual Pulls (at EXITs)	Wet-Pipe Sprinkler LATER L/min per m^2	Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2					>700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	Via exterior door					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area		F8102				
Description		Diesel Driven Fire Pump Room				
Building		Station Water Intake Building				
Fire Zone Dwg		9A.2-203				
Applicable Codes		IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804				
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					3-hour	
Except					Exterior Walls (non-rated)	
Consisting of the following rooms:						
Elevation	Room #	Potential Combustibles	Fire Detection		Fire Suppression	
			Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation No. 2 Diesel Fuel Oil	Smoke	Manual Pulls (at EXITs)	Wet-Pipe Sprinkler 10.2 L/min per m^2 over entire area	Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2					>700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation		To be determined during detailed design				
Radiological release		None, no radiological materials present				
Life safety		To be determined during detailed design				
Manual firefighting		To be determined during detailed design				
Property loss		To be determined during detailed design				
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area	F8103					
Description	Electrical Room					
Building	Station Water Intake Building					
Fire Zone Dwg	9A.2-203					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable insulation Electrical Equipment	Area Wide Ionization	Manual pulls (at EXIT)	Wet-pipe sprinkler 12.2 L/min per m^2 over entire area	Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2					>1400	
Non-sprinkled combustible load limit, MJ/m^2					1400	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	Via EXIT Door					
Property loss	Minor					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area			F8104			
Description			Nitrogen Storage Area			
Building			Nitrogen Storage Area			
Fire Zone Dwg			9A.2-33R			
Applicable Codes			IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 101, 804			
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	LATER	Manual Pulls	None	Hydrants	Fire Extinguisher
Anticipated combustible load, MJ/m^2					LATER	
Non-sprinkled combustible load limit, MJ/m^2					LATER	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation			To be determined during detailed design			
Radiological release			None, no radiological materials present			
Life safety			To be determined during detailed design			
Manual firefighting			To be determined during detailed design			
Property loss			To be determined during detailed design			
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

NAPS
COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F8105					
Description	Hydrogen Storage Area					
Building	Hydrogen Storage Area					
Fire Zone Dwg	9A.2-33R					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Hydrogen	H_2 System Instrumentation	Manual Pull	Yard Hydrants	Fire Extinguishers
Anticipated combustible load, MJ/m^2					>700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

NAPS COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area		F8106				
Description		Oxygen Storage Area				
Building		Oxygen Storage Area				
Fire Zone Dwg		9A.2-33R				
Applicable Codes		IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804				
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	LATER	LATER	LATER	Yard Hydrants	Fire Extinguishers
Anticipated combustible load, MJ/m^2					LATER	
Non-sprinkled combustible load limit, MJ/m^2					LATER	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation		To be determined during detailed design				
Radiological release		None, no radiological materials present				
Life safety		To be determined during detailed design				
Manual firefighting		To be determined during detailed design				
Property loss		To be determined during detailed design				
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

NAPS COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F8107					
Description	Dry Cooling Tower Electrical Building					
Building	Dry Cooling Tower Electrical Building					
Fire Zone Dwg	9A.2-33R					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Electrical Equipment	Area Wide Ionization	Manual Pulls	CO_2 Fire Extinguisher	Hose Rack
Anticipated combustible load, MJ/m^2					< 1400	
Non-sprinkled combustible load limit, MJ/m^2					1400	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

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Table 9A.5-7R Yard (continued)

Fire Area		F8109				
Description		Dry Cooling Tower				
Building		Dry Cooling Tower				
Fire Zone Dwg		9A.2-33R				
Applicable Codes		IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804				
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
		Potential	Fire Detection		Fire Suppression	
Elevation	Room #	Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	LATER	LATER	LATER	Yard Hydrants	Yard Hydrants
Anticipated combustible load, MJ/m^2					LATER	
Non-sprinkled combustible load limit, MJ/m^2					LATER	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation		To be determined during detailed design				
Radiological release		None, no radiological materials present				
Life safety		To be determined during detailed design				
Manual firefighting		To be determined during detailed design				
Property loss		To be determined during detailed design				
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

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Table 9A.5-7R Yard (continued)

Fire Area	F8181					
Description	Hot Machine Shop					
Building	Hot Machine Shop					
Fire Zone Dwg	9A.2-205, 9A.2-206					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification	F-1					
Electrical classification	N/A					
Safety-related divisional equipment or cables	N/A					
Non-safety-related redundant trains or equipment or cables	N/A					
Surrounded by fire barriers rated at	N/A					
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Flammable Solvents Oil	Manual Pulls (at EXITs)	None	Hose Racks	Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2	LATER					
Non-sprinkled combustible load limit, MJ/m^2	LATER					
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment. All safety divisions and both onsite and offsite Power Supplies A and B are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area	F8282					
Description	Electrical Work Area					
Building	Hot Machine Shop					
Fire Zone Dwg	9A.2-205					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Flammable Solvents Oil	Manual Pulls (at EXITs)	None	Hose Racks	Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2					< 1400	
Non-sprinkled combustible load limit, MJ/m^2					1400	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

NAPS COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F8183					
Description	Office Area (First Floor)					
Building	Hot Machine Shop					
Fire Zone Dwg	9A.2-206					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					B	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					3-hour wall against machine shops 2 hours for stairwells and elevator shaft	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation Office Supplies	Smoke	Manual Pulls (at EXITs)	Wet-Pipe Sprinklers LATER L/min per m^2	Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2					>700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

NAPS
COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F8184
Description	Stairwell (South)
Building	Hot Machine Shop
Fire Zone Dwg	9A.2-205, 9A.2-206
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804
Building code occupancy classification	F-1
Electrical classification	N/A
Safety-related divisional equipment or cables	N/A
Non-safety-related redundant trains or equipment or cables	N/A
Surrounded by fire barriers rated at	2-hour
Except	

Consisting of the following rooms:

Elevation	Room #	Potential Combustibles	Fire Detection		Fire Suppression	
			Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation	Area Wide Ionization	Manual Pulls (at EXITs)	Hose Rack	ABC Fire Extinguishers

Anticipated combustible load, MJ/m^2	< 700
Non-sprinkled combustible load limit, MJ/m^2	700

Assuming operation of fire suppression systems, effect of fire upon:

Plant operation	None
Radiological release	None, no radiological materials present
Life safety	To be determined during detailed design
Manual firefighting	To be determined during detailed design
Property loss	To be determined during detailed design

Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:

Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.

NAPS
COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F8185					
Description	Stairwell (North)					
Building	Hot Machine Shop					
Fire Zone Dwg	9A.2-205, 9A.2-206					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 10, 14, 72, 75, 101, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					2 hour	
Except					3-hour against hot machine shop	
Consisting of the following rooms:						
Elevation	Room #	Potential Combustibles	Fire Detection		Fire Suppression	
			Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation	Area Wide Ionization	Manual Pulls (at EXITs)	Hose Racks	ABC Fire Extinguishers
Anticipated combustible load, MJ/m^2					<700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area	F8186					
Description	Elevator					
Building	Hot Machine Shop					
Fire Zone Dwg	9A.2-205, 9A.2-206					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 10, 12, 13, 14, 72, 75, 101, 804; ASME A17.1					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					3-hour wall against machine shops 2 hours for stairwells and elevator shaft	
Except						
Consisting of the following rooms:						
Elevation	Room #	Potential Combustibles	Fire Detection		Fire Suppression	
			Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation Electrical Equipment Class III B Lubricant	Area Wide Ionization	Manual Pulls (at EXITs)	CO_2 Fire Extinguishers ABC Fire Extinguishers (outside elevator at each floor)	Hose Rack
Anticipated combustible load, MJ/m^2					<700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

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COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area		F8187				
Description		HVAC Equipment Room				
Building		Hot Machine Shop				
Fire Zone Dwg		9A.2-205, 9A.2-206				
Applicable Codes		IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804				
Building code occupancy classification					B	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					1-hour	
Except					Exterior Walls	
Consisting of the following rooms:						
Elevation	Room #	Potential Combustibles	Fire Detection		Fire Suppression	
			Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation	Smoke	Manual Pulls (at EXITs)	Wet-Pipe Sprinklers LATER L/min per m^2	Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2					>700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation		None				
Radiological release		None, no radiological materials present				
Life safety		To be determined during detailed design				
Manual firefighting		To be determined during detailed design				
Property loss		To be determined during detailed design				
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

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Table 9A.5-7R Yard (continued)

Fire Area	F8188					
Description	Elevator Maintenance Access					
Building	Hot Machine Shop					
Fire Zone Dwg	9A.2-205, 9A.2-206					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 10, 14, 72, 101, 804; ASME A17.1					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					2 hours	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Cable Insulation Electrical Equipment Class IIIB Lubricants	Area Wide Ionization	Manual Pulls (at EXITs)	CO_2 Fire Extinguisher	ABC Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2					<700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

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COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F8189					
Description	Mechanics Work Area					
Building	Hot Machine Shop					
Fire Zone Dwg	9A.2-206					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					3 hour	
Except					Exterior Walls (non-rated)	
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Flammable Solvents Oil	Manual Pulls (at EXITs)	None	Hose Racks	Fire Extinguishers Yard Hydrants
Anticipated combustible load, MJ/m^2					< 700	
Non-sprinkled combustible load limit, MJ/m^2					700	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.						

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COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

Fire Area	F8200					
Description	Cooling Tower Maintenance Building					
Building	Cooling Tower Maintenance Building					
Fire Zone Dwg	9A.2-33R					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
		Potential	Fire Detection		Fire Suppression	
Elevation	Room #	Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	LATER	Manual Pulls (at EXITs)	None	LATER	LATER
Anticipated combustible load, MJ/m^2					LATER	
Non-sprinkled combustible load limit, MJ/m^2					LATER	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	To be determined during detailed design					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

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Table 9A.5-7R Yard (continued)

Fire Area	F8201					
Description	Hybrid Cooling Tower Electrical Building					
Building	Hybrid Cooling Tower Electrical Building					
Fire Zone Dwg	9A.2-33R					
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804					
Building code occupancy classification					F-1	
Electrical classification					N/A	
Safety-related divisional equipment or cables					N/A	
Non-safety-related redundant trains or equipment or cables					N/A	
Surrounded by fire barriers rated at					N/A	
Except						
Consisting of the following rooms:						
			Fire Detection		Fire Suppression	
Elevation	Room #	Potential Combustibles	Primary	Backup	Primary	Backup
To be determined during detailed design	To be determined during detailed design	Electrical Equipment	Area Wide Ionization	Manual Pulls	CO_2 Fire Extinguishers	Hose Racks
Anticipated combustible load, MJ/m^2					< 1400	
Non-sprinkled combustible load limit, MJ/m^2					1400	
Assuming operation of fire suppression systems, effect of fire upon:						
Plant operation	None					
Radiological release	None, no radiological materials present					
Life safety	To be determined during detailed design					
Manual firefighting	To be determined during detailed design					
Property loss	To be determined during detailed design					
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:						
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment.						

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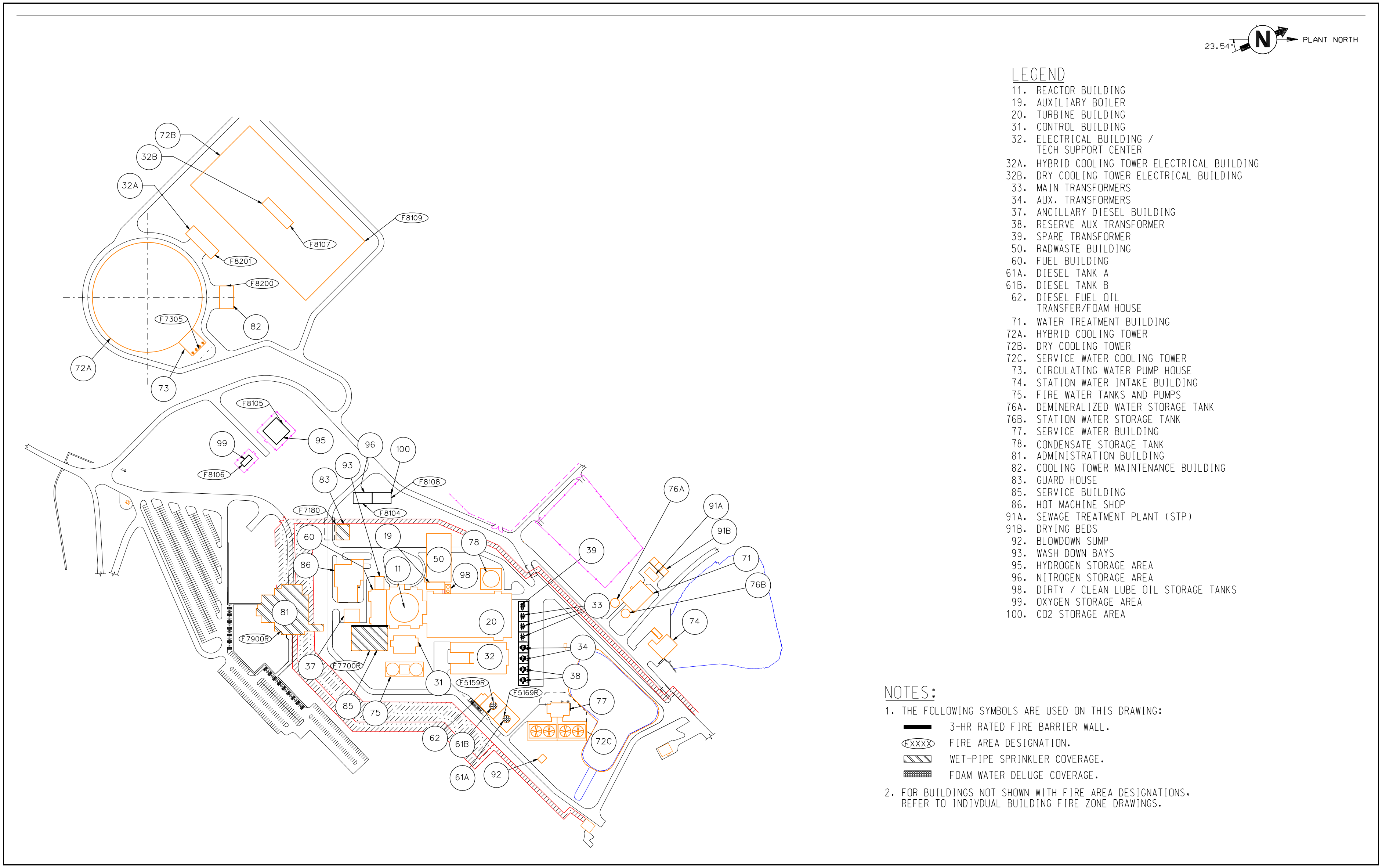
COL 9A.7-2-A

Table 9A.5-7R Yard (continued)

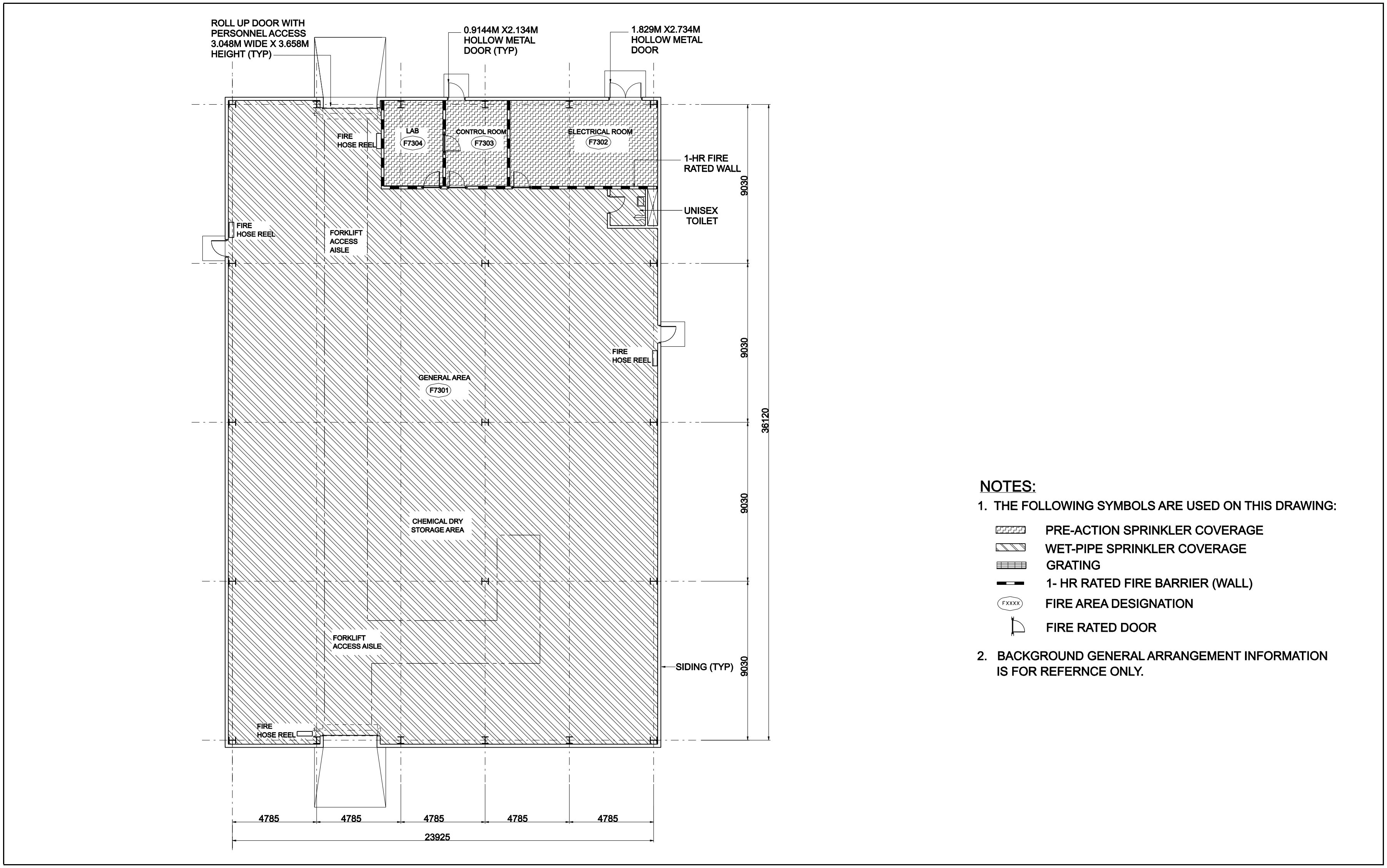
Fire Area	F8283						
Description	Office Area (Second Floor)						
Building	Hot Machine Shop						
Fire Zone Dwg	9A.2-205						
Applicable Codes	IBC; Reg Guide 1.189; NFPA 15, 45, 72, 75, 804						
Building code occupancy classification					B		
Electrical classification					N/A		
Safety-related divisional equipment or cables					N/A		
Non-safety-related redundant trains or equipment or cables					N/A		
Surrounded by fire barriers rated at					3-hour		
Except					Stairwell/Elevator 2 hour Elevator Door 1.5 hour		
Consisting of the following rooms:							
Elevation	Room #	Potential Combustibles	Fire Detection		Fire Suppression		
			Primary	Backup	Primary	Backup	
To be determined during detailed design	To be determined during detailed design	Cable Insulation Office Supplies	Smoke	Manual Pulls (at EXITs)	Wet-Pipe Sprinklers LATER L/min per m^2	Fire Extinguishers	
Anticipated combustible load, MJ/m^2					>700		
Non-sprinkled combustible load limit, MJ/m^2					700		
Assuming operation of fire suppression systems, effect of fire upon:							
Plant operation	None						
Radiological release	None, no radiological materials present						
Life safety	To be determined during detailed design						
Manual firefighting	To be determined during detailed design						
Property loss	To be determined during detailed design						
Assuming all fire suppression systems inoperable, effect of design basis fire on safe shutdown:							
Complete burnout of all equipment and cables within this fire area affects no safety-related or safe shutdown divisional equipment; all safety divisions and both on-site and off-site Power Supplies A and B are unaffected by fire and are operable.							

Figure 9A.2-33R Site Fire Protection Zone ESBWR Plot Plan



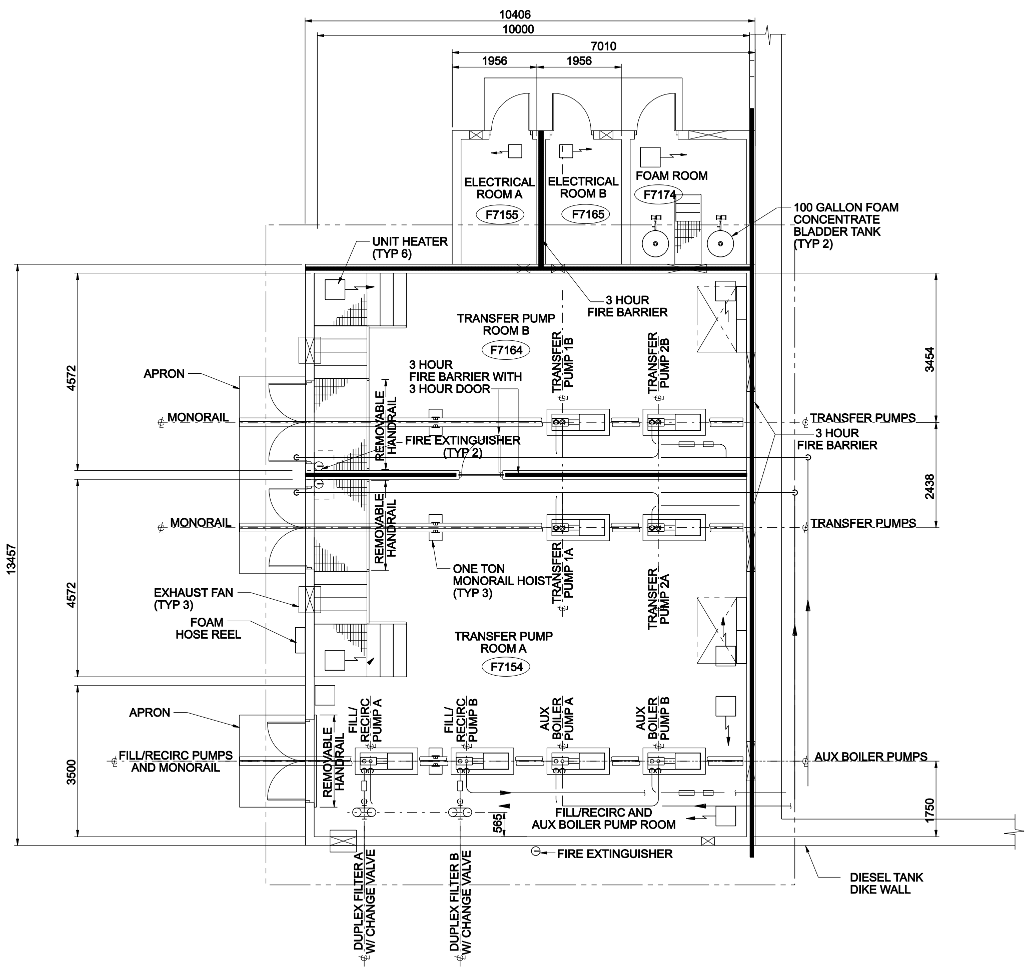
STD COL 9A.7-1-A

Figure 9A.2-201 Fire Zones - Water Treatment Building



STD COL 9A.7-1-A

Figure 9A.2-202 Fire Zones - Diesel Fuel Oil Transfer/Foam House

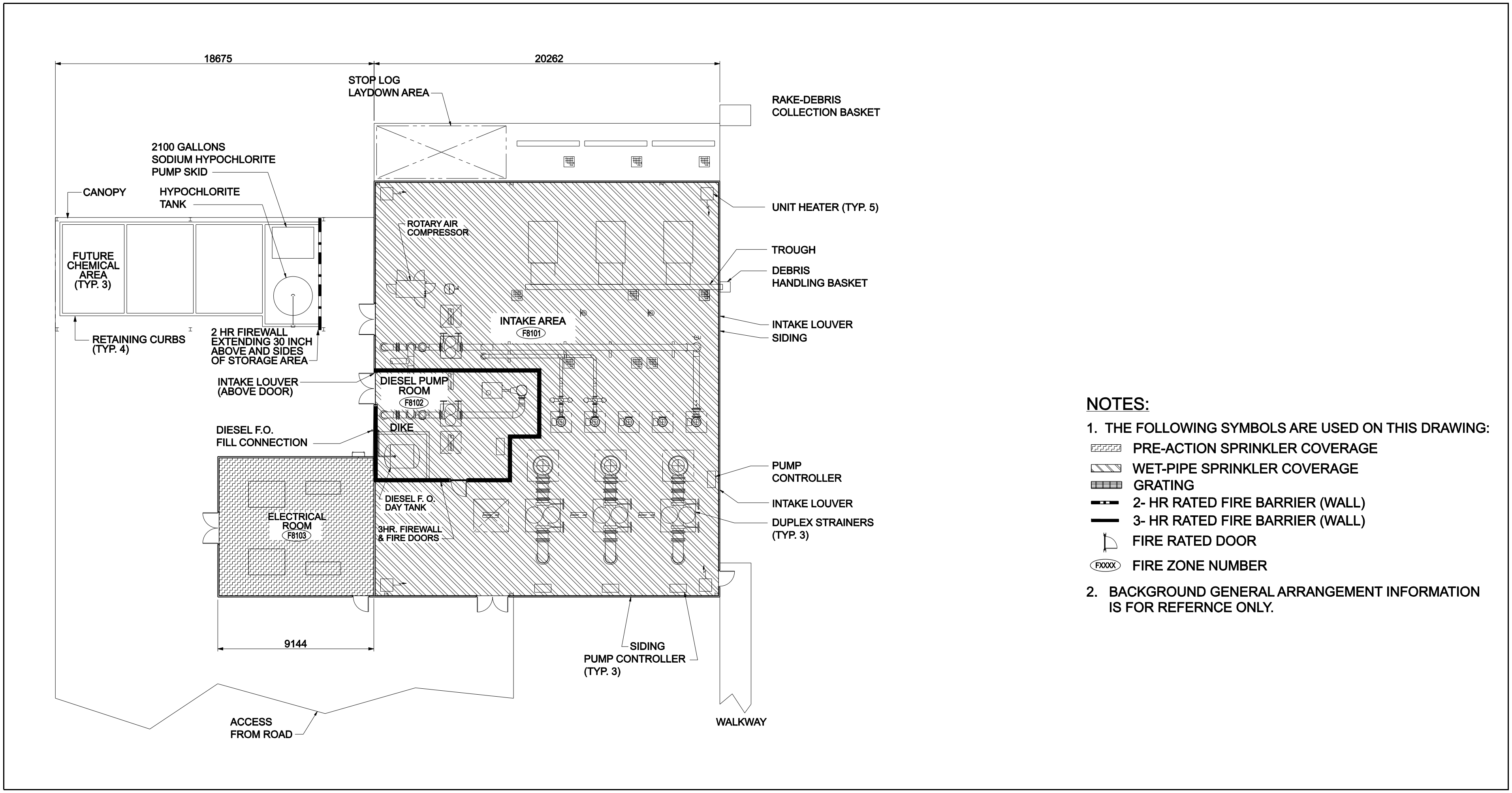


NOTES:

1. THE FOLLOWING SYMBOLS ARE USED ON THIS DRAWING:
   - GRATING
   - 1- HR RATED FIRE BARRIER (WALL)
   - 2- HR RATED FIRE BARRIER (WALL)
   - 3- HR RATED FIRE BARRIER (WALL)
   - FIRE RATED DOOR
   - FXXXX FIRE AREA DESIGNATION
2. BACKGROUND GENERAL ARRANGEMENT INFORMATION IS FOR REFERNCE ONLY.

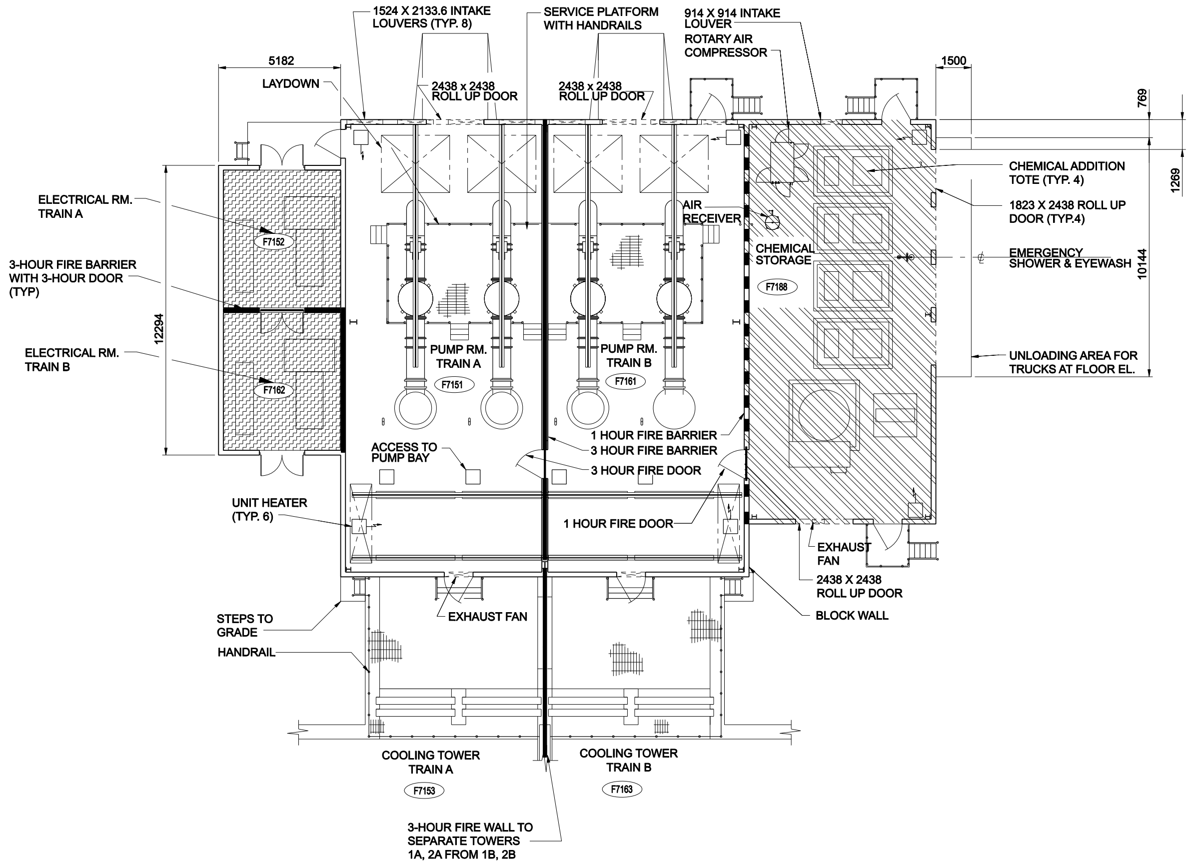
STD COL 9A.7-1-A

Figure 9A.2-203 Fire Zones - Station Water Intake Building



STD COL 9A.7-1-A

Figure 9A.2-204 Fire Zones - Service Water Building

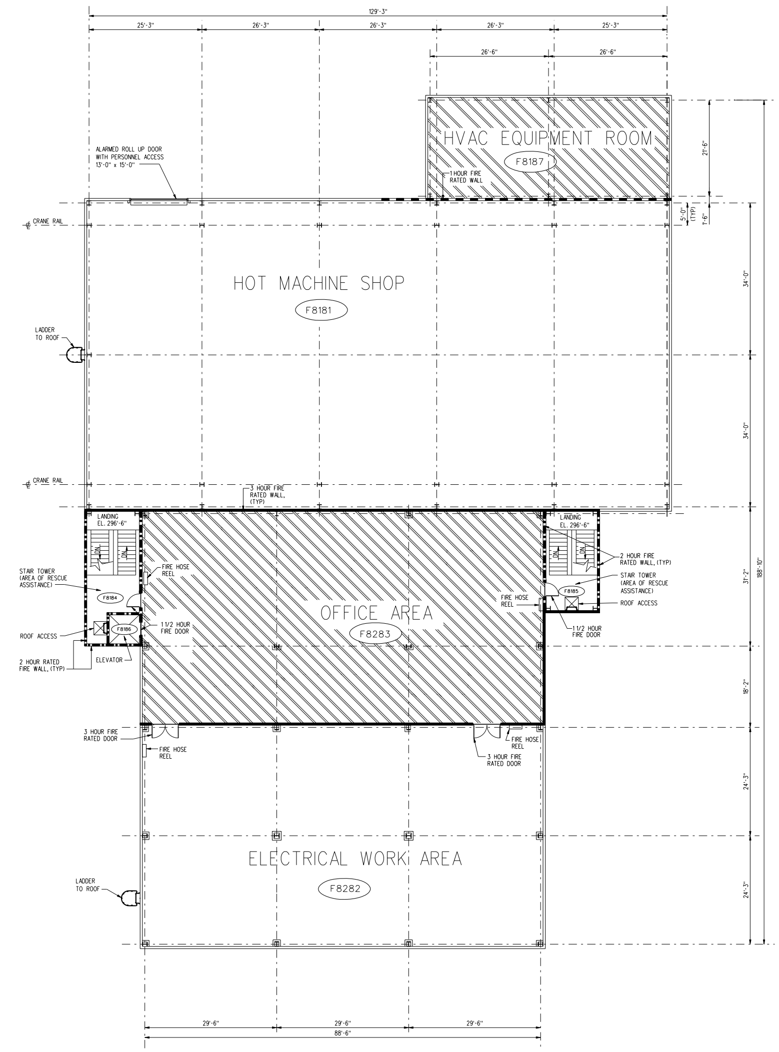


NOTES:

1. THE FOLLOWING SYMBOLS ARE USED ON THIS DRAWING:
   - PRE-ACTION SPRINKLER COVERAGE
   - WET-PIPE SPRINKLER COVERAGE
   - GRATING
   - 1- HR RATED FIRE BARRIER (WALL)
   - 3- HR RATED FIRE BARRIER (WALL)
   - FIRE RATED DOOR
   - FXXXX FIRE ZONE NUMBER
2. BACKGROUND GENERAL ARRANGEMENT INFORMATION IS FOR REFERNCE ONLY.

STD COL 9A.7-1-A

Figure 9A.2-205 Fire Zones - Hot Machine Shop Second Floor

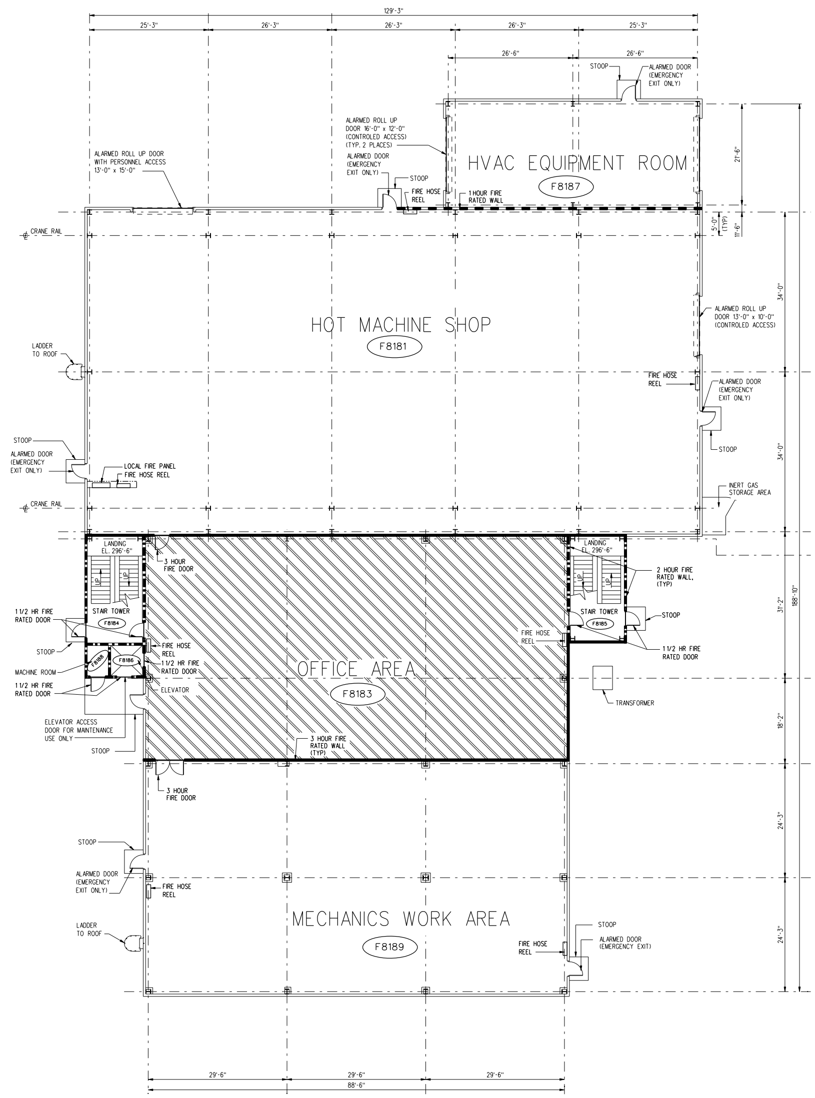


NOTES:

1. THE FOLLOWING SYMBOLS ARE USED ON THIS DRAWING:
   - WET-PIPE SPRINKLER COVERAGE
   - GRATING
   - 1- HR RATED FIRE BARRIER (WALL)
   - 2- HR RATED FIRE BARRIER (WALL)
   - 3- HR RATED FIRE BARRIER (WALL)
   - FIRE AREA DESIGNATION
   - FIRE RATED DOOR
2. BACKGROUND GENERAL ARRANGEMENT INFORMATION IS FOR REFERNCE ONLY.

STD COL 9A.7-1-A

Figure 9A.2-206 Fire Zones - Hot Machine Shop First Floor



NOTES:

1. THE FOLLOWING SYMBOLS ARE USED ON THIS DRAWING:
   - WET-PIPE SPRINKLER COVERAGE
   - GRATING
   - 1- HR RATED FIRE BARRIER (WALL)
   - 2- HR RATED FIRE BARRIER (WALL)
   - 3- HR RATED FIRE BARRIER (WALL)
   - FXXXX FIRE AREA DESIGNATION
   - FIRE RATED DOOR
2. BACKGROUND GENERAL ARRANGEMENT INFORMATION IS FOR REFERNCE ONLY.

Appendix 9B Summary of Analysis Supporting Fire Protection Design Requirements

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

Chapter 10 Steam and Power Conversion System

10.1 Summary Description

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

10.2 Turbine Generator

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

10.2.3.4 Turbine Design

Add the following at the beginning of this section.

STD SUP 10.2-1

The General Electric Company manufactures the turbine and generator. The model N3R-6F52 turbine is from General Electric's N series nuclear steam turbines.

10.2.3.6 Inservice Maintenance and Inspection of Turbine Rotors

Replace the last paragraph with the following.

STD COL 10.2-1-A

The turbine maintenance and inspection program that supports the Original Equipment Manufacturer's turbine missile generation probability calculation is described in DCD Sections 10.2.2.7, 10.2.3.5, 10.2.3.6, and 10.2.3.7. The associated turbine maintenance and inspection frequencies will be established upon completion of the bounding missile probability analysis. This analysis will be completed in the second quarter of 2009 and the FSAR will be revised to incorporate the maintenance and inspection frequencies as part of a subsequent FSAR update.

10.2.3.8 Turbine Missile Probability Analysis

Replace the last paragraph with the following.

STD COL 10.2-2-A

The probability of turbine missile generation will be calculated based on bounding material property values until actual material test specimens are available for testing. The bounding analysis will be completed in the second quarter of 2009 and the FSAR will be revised to reflect this analysis as part of a subsequent FSAR update.

10.2.5 COL Information

10.2-1-A Turbine Maintenance and Inspection Program

STD COL 10.2-1-A This COL Item is addressed in Section 10.2.3.6

10.2-2-A Turbine Missile Probability Analysis

STD COL 10.2-2-A This COL Item is addressed in Section 10.2.3.8.

10.3 Turbine Main Steam System

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

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.

10.4.5.2.1 General Description

Replace the text with the following.

NAPS CDI The CIRC is depicted in Figures 10.4-201 through 10.4-203. The CIRC consists of the following components:

- Condenser water boxes, piping, and valves
- Condenser tube cleaning equipment
- Water box drain subsystem
- Four 25 percent capacity pumps and pump discharge valves
- A removable assembly of coarse and fine screens that separate the pump forebay (suction) from the hybrid cooling tower basin
- An array of dry, mechanical draft cooling tower cells arranged in banks
- One combination (hybrid) wet/dry, mechanical draft cooling tower

Table 10.4-3R includes the temperature range of the water delivered by the CIRC pumps to the main condenser.

The CIRC water is normally circulated by four motor-driven pumps through the condenser and back to the cooling towers. Depending on ambient conditions, system configuration, and heat load, one CIRC pump may be taken out of operation with the flow of the remaining three CIRC pumps providing sufficient water for condenser heat removal.

The four pumps are arranged in parallel. Discharge lines combine into two parallel main circulating water supply lines to the main condenser. Each main circulating water supply line connects to a low pressure condenser inlet water box.

Two interconnecting lines are provided between the two main circulating water supply lines. The first interconnecting line is located near the discharge of the circulating water pumps and is used for flow balancing. The second interconnecting line is near the location where the CIRC pipes enter the turbine building and is used as a blowdown point. A motor operated isolation valve is provided on the flow balancing line. Two motor operated valves are located on the blowdown cross-connect line, one on either side of the blowdown line. These valves allow operation of the CIRC with one main circulating water supply line out of service.

The discharge of each pump is fitted with a remotely operated valve. This arrangement permits isolation and maintenance of any one pump while the others remain in operation and minimizes the backward flow through an out-of-service pump.

The CIRC and condenser are designed to permit isolation of half of the three series connected tube bundles to permit repair of leaks and cleaning of water boxes while operating at reduced power.

The CIRC includes water box vents to help fill the condenser water boxes during startup and remove accumulated air and other gases from the water boxes during normal operation.

The CIRC system incorporates design provisions that minimize the effect of hydraulic transients upon the functional capability and the integrity of the system components. These design features include slow-stroke motor-operated valves (MOVs), air release valves to fill and keep the system full, vacuum release valves that minimize pressure transients, valve control and interlock features that ensure correct valve line-up prior to pump start, and discharge isolation valves that open and close with pump start and stop signals.

Circulating water chemistry is maintained by the Chemical Storage and Transfer System and with blowdown. Circulating water chemical equipment injects the required chemicals into the circulating water pump bay before entering the circulating water pumps.

10.4.5.2.2 Component Description

Replace the last paragraph with the following.

NAPS CDI

Table 10.4-3R provides reference parameters for the major components of the CIRC.

10.4.5.2.2.1 CIRC Chemical Injection

Circulating water chemistry is maintained by the Chemical Storage and Transfer System. Chemical feed equipment injects the required chemicals into the circulating water at the pump bay before water enters the circulating water pumps.

Chemical injection maintains a non-corrosive, non-scale-forming condition and limits the biological film formation that reduces the heat transfer rate in the condenser and cooling towers.

Plant chemistry specifies the required chemicals used within the system. The chemicals can be divided into five categories based upon function: biocide, algaecide, pH adjuster, corrosion inhibitor, and scale inhibitor. The pH adjuster, corrosion inhibitor, and scale inhibitor are metered into the system continuously or as required to maintain proper concentrations. Biocide application frequency may vary with seasons. Algaecide is applied, as necessary, to control algae formation in the cooling towers. Chemicals that are injected in the CIRC include sodium hypochlorite, acid, bromide, dispersants, and non-oxidizing biocides.

Circulating water chemistry is also controlled as required with blowdown.

Chemicals selected are compatible with selected materials or components used in the CIRC.

10.4.5.2.3 System Operation

Add the following at the end of this section.

NAPS CDI

The four circulating water pumps take suction from the pump forebay and circulate the water through the main condenser. Circulating water returns through the condenser discharge to the cooling towers. The operating configuration of the cooling towers and CIRC is modified depending on desired configuration, heat load, and ambient conditions.

Circulating water discharged from the condenser first passes through the dry cooling tower arrays where sensible heat is removed. The water then passes through the dry section of the hybrid tower, where additional sensible heat is removed prior to entering the wet section of the hybrid tower. In the wet section, the water is distributed through nozzles in the hybrid cooling tower's distribution headers. The water then falls through film-type fill material to the basin beneath the tower. In the process, the water rejects additional heat to the atmosphere through direct contact with the air and evaporation of a small amount of water.

Provisions are made to vary the operation of the CIRC and cooling towers during specific ambient conditions such as hot and cold weather and in response to specific environmental conditions such as periods of low water level in Lake Anna. Various configurations are utilized where select mechanical draft fans are started, operated at reduced speed, or stopped, select portions or all of the NPHS is bypassed, and condenser halves are isolated. These alternate and transitional configurations are utilized to provide benefits such as freeze protection, water conservation, energy conservation, plume minimization, and isolation of portions of the CIRC and other systems for maintenance.

Selected components may be taken out of service during power operation. These alternate configurations normally change plant thermal performance. In some configurations, reactor power reduction may be required to avoid a turbine trip on decreasing condenser vacuum.

The SWS supplies makeup water to the circulating water pump forebay to replace water losses due to evaporation, drift, and blowdown. Blowdown from the CIRC is taken from the cross-connect near the turbine building. The blowdown flow is discharged to the plant discharge canal at a maximum of 37.8ºC (100ºF).

A condenser tube cleaning subsystem cleans the circulating water side of the main condenser tubes.

Leakage of condensate from the main condenser into the CIRC via a condenser tube leak is not likely during power operation, since the CIRC normally operates at a greater pressure than the shell (condensate) side of the condenser. Analysis of routine CIRC cooling tower grab samples will detect events that could lead to unmonitored, uncontrolled radioactive releases to the environment. This provides the action required by NRC Inspection and Enforcement Bulletin No. 80-10.

10.4.5.5 Instrumentation Applications

Insert the following between the fourth and fifth paragraphs.

NAPS CDI

Level instrumentation provided in the circulating water pump forebay controls makeup flow from the SWS to the pump forebay via the N-DCIS. Level instrumentation in the pump forebay initiates alarms in the main control room on abnormally low or high water level.

Pressure indication is provided on the circulating water pump discharge. Differential pressure instrumentation is provided across the inlet and outlet to the condenser and is used to determine the frequency of operating the condenser tube cleaning system.

Local grab samples are used to periodically test the circulating water quality.

Replace the last paragraph with the following.

The temperature in each condenser cooling water supply line is indicated in the MCR. Based on these indications, warm water recirculation is controlled to maintain a minimum inlet temperature of approximately 0°C (32°F).

10.4.5.6 Flood Protection

Add the following at the end of this section.

NAPS CDI

Failure of a pipe or component in the CIRC hybrid cooling tower or elsewhere in CIRC in the yard would not have an adverse impact on the intended design functions of safety-related SSCs.

For the hybrid cooling tower, the largest components are the two vertical large-bore CIRC pipes that connect to the hybrid cooling tower's distribution headers. It is conservatively assumed that these large CIRC underground pipes surface outside the confines of the hybrid cooling tower basin.

A postulated rupture of one of these pipes would result in water flow in the area of the yard with the cooling towers. The yard in this area slopes to the west. Water discharged from such a break would flow down to the drainage ditch along the west side of the cooling tower area and drain away from Unit 3 toward Lake Anna.

Depending on the size and orientation of the break, some discharging water may flow eastward toward a drainage ditch along the east side of the cooling tower area or toward the access road leading to Unit 3. Water reaching the access road would flow into the ditches along the plant access road. The flow-rate in the ditches past the power block area would be less than that considered for the local PMP event. Therefore, safety-related SSCs would not be subjected to flooding as a result of a failure of the largest hybrid cooling tower component.

The failure of this vertical large-bore CIRC pipe bounds other failures of piping and components in the CIRC. The remainder of the system is either underground or has a smaller diameter. Failures of these underground and smaller diameter components would have lower flow-rates than a postulated failure of a vertical, above-ground, large-bore CIRC pipe. Also, flow from such failures would be either in the cooling tower area or toward the plant access road ditches and to either the storm water basin or the make-up water intake area.

Failure of the CIRC hybrid cooling tower basin has also been considered. Because the basin is an in-ground structure, the maximum water level elevation in the hybrid cooling tower basin is lower than the elevations of the surrounding areas. This design and the selected location ensure that failure of the basin results in no water discharge to the surface. However, should any discharge occur, the water would flow toward the lake rather than toward the plant.

10.4.5.8 **Normal Power Heat Sink**

Replace the text with the following.

NAPS CDI

The cooling tower arrangement includes a dry cooling tower array and a round, wet/dry (hybrid) cooling tower that may operate independently or in series. The towers may be bypassed or partially or fully utilized as required, depending on desired operating configuration, heat load, and ambient conditions.

The dry tower array is arranged in rectangular banks of multiple cells. Each cell includes air cooled heat exchange surfaces, a motor-driven mechanical draft fan, and inlet and outlet isolation valves. The round, hybrid cooling tower includes a dry upper section and a wet lower section. Both the wet and dry sections of the hybrid tower include mechanical draft fans to provide air flow. The combination of dry and hybrid cooling tower arrangements supports a condenser maximum cold water temperature of 35ºC (100ºF).

Both the dry and hybrid cooling towers are located at least a distance equal to their height away from any seismic Category 1 or 2 structures. Thus, if there were any structural failure of the cooling towers, no Seismic Category 1 or 2 structures or any safety-related systems or components would be affected or damaged.

Both the dry and hybrid cooling towers have multiple fans with associated motors, couplings, and gearboxes. The fans rotate at relatively slow speeds and the fan blades are made of relatively low-density material. A failure of a fan could result in the generation of missiles. However, due to the site arrangement and construction of the respective towers, any damage would be confined to the cooling towers. Therefore, there would be no damage to any Seismic Category 1 or 2 structures or any safety-related systems or components.

10.4.6.3 Evaluation

Replace the second sentence in the third paragraph with the following.

STD COL 10.4-1-A A table summarizing the manufacturer's recommended threshold values of key chemistry parameters and associated operator actions is provided as Table 10.4-201.

10.4.10 COL Information

10.4-1-A Leakage (of Circulating Water Into the Condenser)

STD COL 10.4-1-A This COL Item is addressed in Section 10.4.6.3.

STD COL 10.4-1-A

Table 10.4-201 Recommended Water Quality and Action Levels

Reactor Water Quality-Power Operation

Control Parameter	Action Levels 0	1	2	3
Conductivity, µS/cm at 25ºC*	≤ 0.100	> 0.300	> 1	≥ 2
Chloride, ppb	≤ 0.3	> 5	> 50	≥ 200
Silica, ppb	≤ 200	> 500	N/A	N/A
Sulfate, ppb	≤ 2	> 5	> 50	≥ 200

Feedwater Quality—Power Operation*

Control Parameter	Action Levels 0	1	2
Conductivity, µS/cm at 25ºC**	< 0.057	> 0.065	> 0.100
Dissolved Oxygen, ppb as O_2**	30-50	< 20 or > 200	N/A

* Value depends on Hydrogen Water Chemistry System operation
** Applicable when Reactor Power >10%
*** Also Condensate Purification System Effluent

Action Level 0: Target Value. The parameter may be outside the Action Level 0 value and not in Action Level 1, 2, or 3. In this case, efforts should be made to return the parameter to the Action Level 0 value.

Action Level 1: Lowest Severity. The parameter should be brought below this value within 96 hours. A technical review should be performed to determine the appropriate response.

Action Level 2: Moderate Severity. If the parameter is not reduced below this level within 24 hours, an orderly shutdown should be initiated.

Action Level 3: Highest Severity. If the parameter is not reduced below this level within 6 hours, an orderly shutdown should be initiated.

NAPS CDI

Table 10.4-3R Circulating Water System

Parameter	Value
Circulating Water Pumps	
Number of pumps	4
Pump type	Vertical, wet pit, turbine
Unit flow capacity**, m^3/hr (gpm)	Approx. 38,500 (169,600)
Driver Type	Electric motor
Normal Power Heat Sink	
Normal Heat Removal Duty @35ºC (95ºF) CIRC Supply Temperature, MW (BTU/hr)	2930 (1.00×10^{10})
Dry Cooling Tower Array	
Array Length*, m (ft)	223 (731)
Array Width*, m (ft)	114 (375)
Array Height*, m (ft)	20 (65)
Wet/Dry (Hybrid) Cooling Tower	
Outside Base Diameter*, m (ft)	150 (492)
Height*, m (ft)	55 (180)
Operating Temperatures	
Temperature range of water delivered to the main condenser, ºC (ºF)	0*** to 37.8 (32 to 100)
CIRC temperature for rated turbine performance, ºC (ºF)	30 (86)
Maximum CIRC temperature to accommodate the bypass flow resulting from a turbine trip, 100% load reject, or island mode, in conjunction with the power reduction resulting from SRI/SCRRI function, ºC (ºF)	35.6 (96)

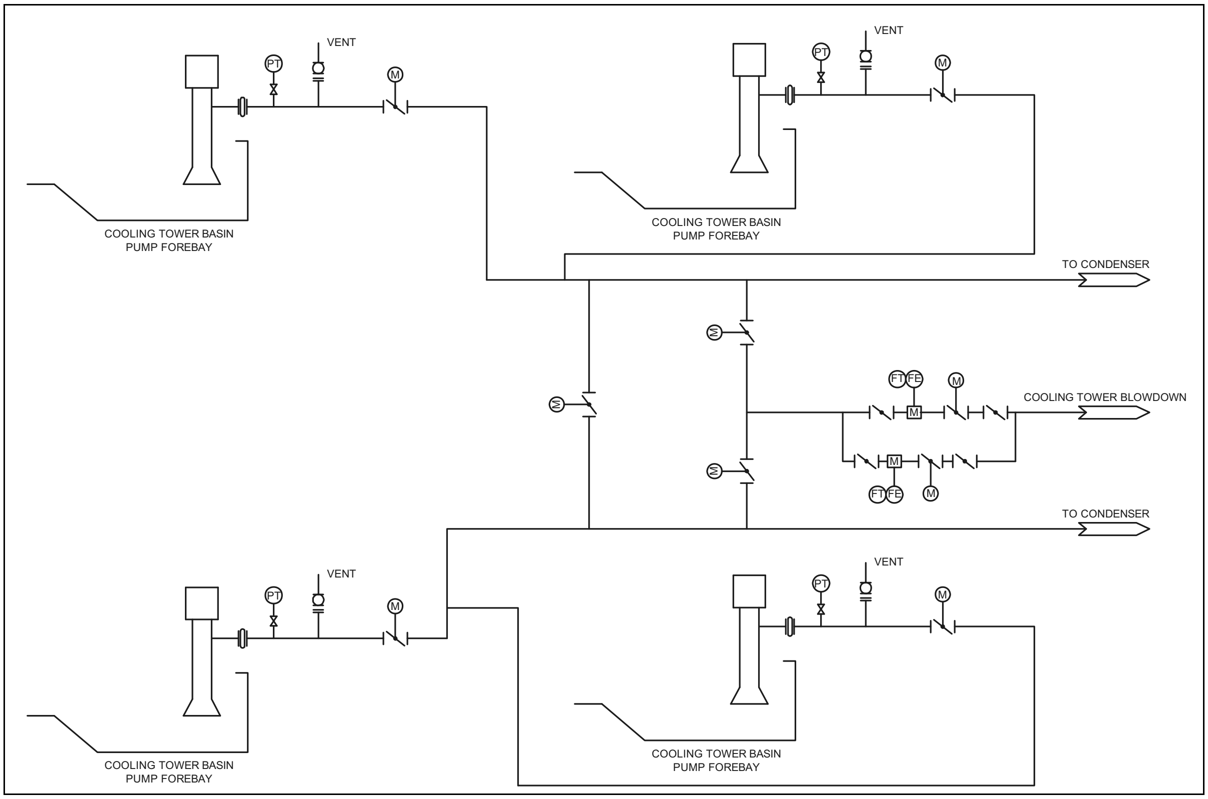
* Cooling tower dimensions and specifications are approximate.

** This capacity is for condenser cooling and blowdown at design temperature of 37.8ºC (100ºF).

*** If the Normal Power Heat Sink does not maintain temperatures above the minimum temperature, then the minimum temperature is maintained by warm water recirculation and cooling tower bypass.

NAPS CDI

Figure 10.4-201 Circulating Water Pumps



NAPS CDI

Figure 10.4-202 Dry Cooling Tower Array

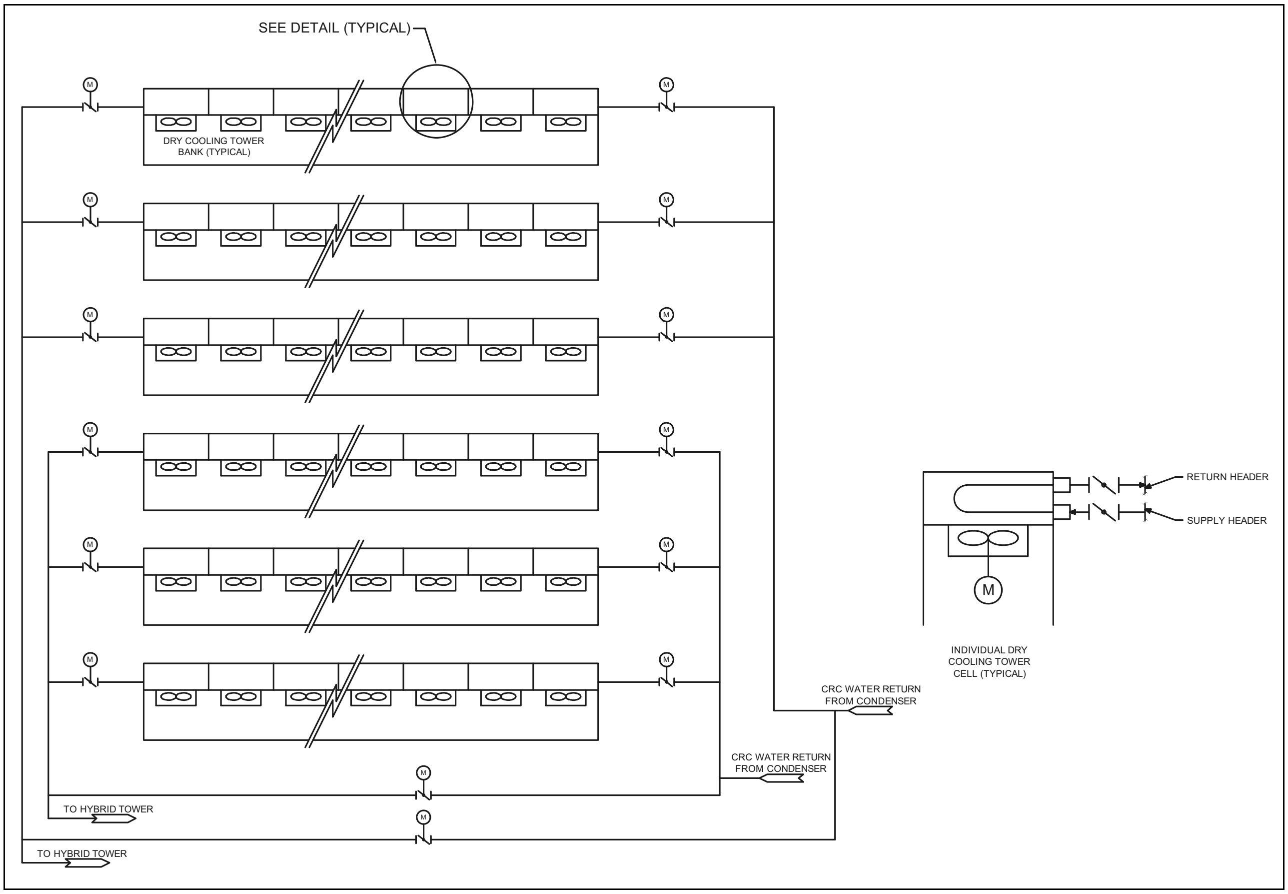
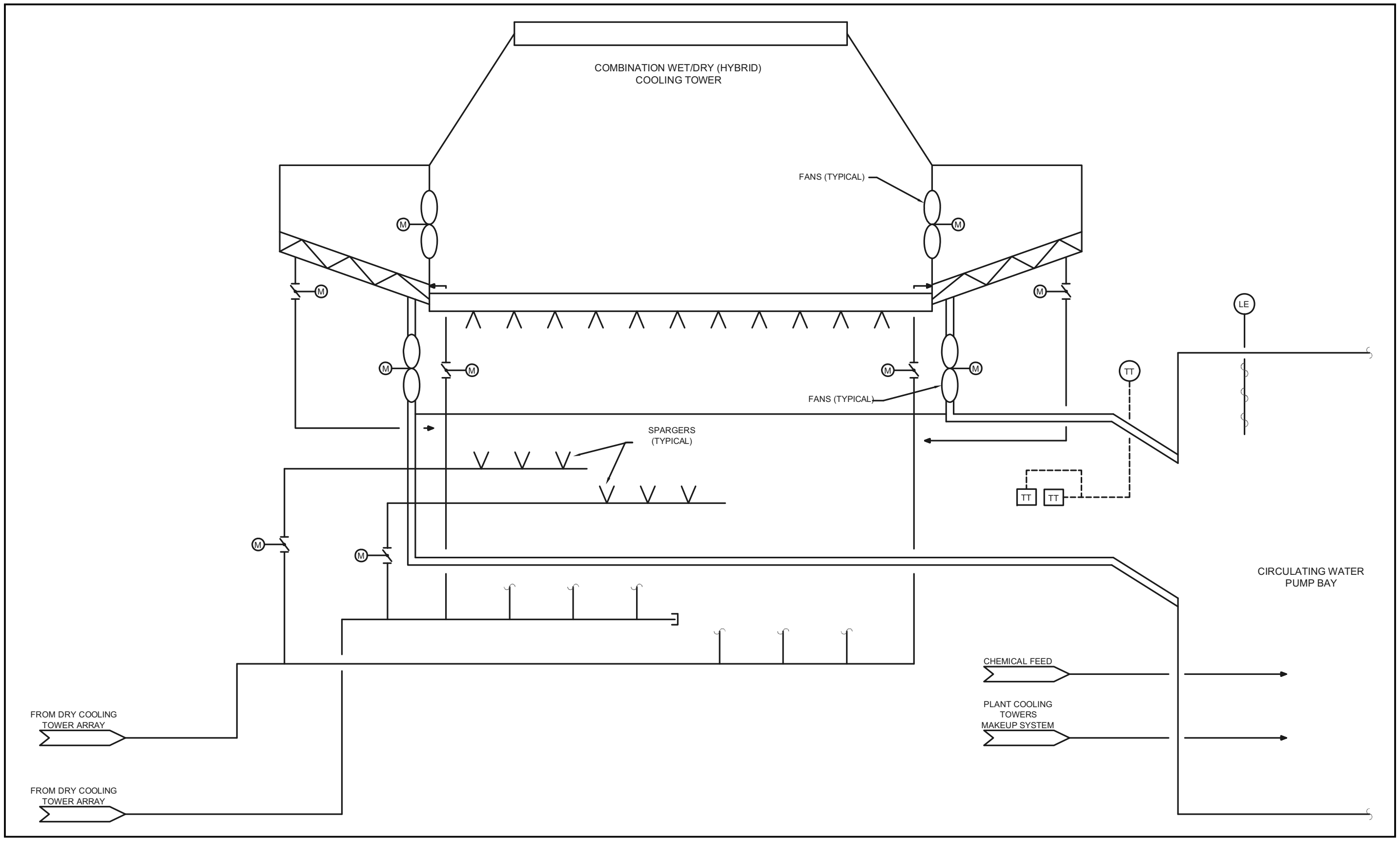


Figure 10.4-203 Hybrid Cooling Tower



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 System

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

11.2.1 Design Basis

Safety Design Bases

Add the following at the end of this section.

NAPS SUP 11.2-1

RG 1.110 methodology was applied to satisfy the cost-benefit analysis requirements of 10 CFR 50, Appendix I, Section II.D, for the system augments compatible with BWR plant design features. Cost parameters used to calculate the Total Annual Cost (TAC) for each applicable radwaste treatment system augment listed in RG 1.110 are taken without exception from RG 1.110, Appendix A. These costs are 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). Other cost parameters used to determine TAC are as follows:

- Capital Recovery Factor (CRF) - Obtained from RG 1.110, Table A-6, this factor reflects the cost of money for capital expenditures. A cost-of-money value of 7 percent per year is assumed in this analysis, consistent with “Guidelines and Discount Rates for Benefit-Cost Analysis of Federal Programs” (OMB Circular A-94) (Reference 11.2-202). Based on a 30-year service life, Table A-6 gives a CRF of 0.0806.
- Indirect Cost Factor (ICF) - Obtained from RG 1.110, Table A-5, this factor takes into account whether the radwaste system is unitized or shared (in the case of a multi-unit site). Because this is a single ESBWR unit site, this analysis is for a single unit, which gives an ICF of 1.75.

- Labor Cost Correction Factor (LCCF) - Obtained from RG 1.110, Table A-4, this factor takes into account the relative labor cost differences among geographical regions. A factor of 1 (the lowest value) is assumed in this analysis.

A value of $1,000 per person-rem is prescribed in 10 CFR 50, Appendix I.

If it is conservatively assumed that each radwaste treatment system augment is a “perfect” technology that reduces the effluent dose by 100 percent, the annual cost of the augment can be determined and the lowest annual cost can be considered a threshold value. The lowest-cost option for augments is a 20 gpm cartridge filter at $11,380 per year, which yields a threshold value of 11.38 person-rem whole body or thyroid dose from liquid effluents.

The total body and thyroid doses to the population for the liquid effluents from Unit 3 are given in Section 12.2.2.4.2. None of the augments provided in RG 1.110 is found to be cost beneficial in reducing the annual population doses of 1.0 person-rem total body and 0.69 person-rem thyroid.

The lowest cost liquid radwaste augment is $11,380/year. Implementing this augment would cost $11,380 per person-rem in total body dose reduction, which exceeds the $1,000 per total body person-rem criterion prescribed in 10 CFR 50, Appendix I. Also, implementing this augment would cost $16,500 per person-rem in thyroid dose reduction which exceeds the $1,000 per person-thyroid-rem criterion prescribed in 10 CFR 50, Appendix I. Therefore, even this lowest-cost augment is not cost beneficial.

11.2.2.3 Detailed System Component Description

11.2.2.3.3 Processing Systems

Replace the first two paragraphs with the following.

STD COL 11.2-1-A

Specific equipment connection configuration and plant sampling procedures are used to implement the guidance in Inspection and Enforcement (IE) Bulletin 80-10 (DCD Reference 11.2-10). The permanent and mobile/portable non-radioactive systems, which are connected to radioactive or potentially radioactive portions of process LWMS, are protected from contamination with an arrangement of double check valves in each line. The configuration of each line is also equipped with a tell-tale connection, which permits periodic checks to confirm the integrity of the line and its check valve arrangement. Plant procedures describe sampling of non-radioactive systems that could become contaminated by cross-connection with systems that contain radioactive material. In accordance with the guidance in RG 1.109, exposure pathways that may arise due to unique conditions are considered for incorporation into the plant-specific ODCM if they are likely to contribute significantly to the total dose.

STD COL 11.2-2-A

Section 12.6 discusses how ESBWR design features and procedures for operation will minimize contamination of the facility and environment, facilitate decommissioning, and minimize the generation of radioactive wastes, in compliance with 10 CFR 20.1406. Section 13.5 describes the requirement for procedures for operation of radioactive waste processing system. Operating procedures for LWMS process systems required by Section 12.4, Section 12.5, and Section 13.5 address the requirements of 10 CFR 20.1406.

11.2.6 COL Information

11.2-1-A Implementation of IE Bulletin 80-10

STD COL 11.2-1-A

This COL item is addressed in Section 11.2.2.3.

11.2-2-A Implementation of Part 20.1406

STD COL 11.2-2-A

This COL item is addressed in Section 11.2.2.3.

11.2.7 References

11.2-201 [Deleted]

11.2-202 OMB Circular A-94, “Guidelines and Discount Rates for Benefit-Cost Analysis of Federal Programs,” October 29, 1992, Office of Management and Budget.

11.3 Gaseous Waste Management System

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

11.3.1 Design Basis

Add the following at the end of this section.

NAPS ESP COL 11.1-1

The methodology for performing cost-benefit analysis for the radwaste system is presented in Section 11.2.1.

The annual costs for augments for the gaseous radwaste treatment system were determined and the lowest annual cost was considered a threshold value. The lowest-cost option for a gaseous radwaste treatment system augment that applies to BWRs is the 1000 cfm Charcoal/HEPA Filtration System at $7,960 per year, which yields a threshold value of 7.96 person-rem whole body or thyroid from gaseous effluents for BWRs.

As shown in Table 12.2-204, the Unit 3 annual whole body dose from gaseous effluents is 7.7 person-rem/yr, which is below the 7.96 person-rem/yr threshold value. Based on this comparison, no gaseous radwaste treatment system augment is cost-beneficial in reducing annual whole body dose and the cost-benefit analysis demonstrates compliance with 10 CFR 50, Appendix I, Section II.D, for this type of dose.

As shown in the table below, the Unit 3 thyroid dose from gaseous effluents is 28 person-rem/yr, which exceeds the 7.96 person-rem/yr threshold value for a BWR. Because the Unit 3 estimate exceeds this threshold value, further analysis is provided below.

Source	Thyroid Dose (person-rem/year)	% of Total
Iodines	20	72.9
Particulates	0.65	2.3
Noble gases	1.5	5.4
C-14	5.1	18.6
H-3	0.20	0.7
Total	28	100.0

The cost-benefit analysis described in Section 11.2.1 is based on RG 1.110, which provides the gaseous radwaste augments applicable to a BWR to be considered for Unit 3. Based on the estimated 28 person-rem/year thyroid dose, those augments with a TAC less than

$28,000 are considered below. In some cases, the system augments less than $28,000 per year have insufficient capacity. System augments with greater capacities were considered but eliminated because they had TAC values greater than $28,000. The gaseous radwaste system augments in RG 1.110 applicable to a BWR were considered.

15,000 and 30,000 cfm HEPA Filtration System (If in Auxiliary Building)

For Unit 3, the gaseous effluent releases “from the Auxiliary Building” were considered as follows because an ESBWR does not have an Auxiliary Building. Two ventilation systems that service contaminated air in the Reactor Building are combined: the Contaminated Area HVAC Subsystem (CONAVS) and the Refueling and Pool Area HVAC Subsystem (REPAVS). Per DCD Figure 9.4-10, the normal flow through the CONAVS exhaust fan is 19,950 l/sec (42,272 cfm). Per DCD Figure 9.4-11, the normal flow through the REPAVS exhaust fan is 32,050 l/sec (67,910 cfm). In both cases, the normal flow rates exceed the proposed 7079 l/sec (15,000 cfm) HEPA filtration system. Therefore, this augment is not effective for Unit 3 and is eliminated from further consideration. The 14,158 l/sec (30,000 cfm) Charcoal/HEPA Filtration System is also not effective and with a TAC of $56,330/yr, also exceeds the $28,000/yr TAC threshold.

15,000 and 30,000 cfm HEPA Filtration System (If in Turbine Building)

The Turbine Building HVAC System (TBVS) services the Turbine Building. DCD Figure 9.4-8 shows that the Turbine Building exhaust goes through the Turbine Building Air Exhaust Subsystem (TBE). Per DCD Table 9.4-15, the 100 percent capacity flow through TBE is 52,800 l/sec (111,877 cfm). Based on this design capacity, it is assumed that the normal flow exceeds 7079 l/sec (15,000 cfm), which is 13 percent of the design capacity. Therefore, this augment is not effective for Unit 3 and is eliminated from further consideration. The 14,158 l/sec (30,000 cfm) Charcoal/HEPA Filtration System is also not effective and with a TAC of $54,220/yr, also exceeds the $28,000/yr TAC threshold.

3-Ton Charcoal Adsorber

Per DCD Table 11.3-1, the total mass of charcoal in the offgas system is 237 metric tons (523,000 lb), or approximately 262 tons. Addition of a 2.7 metric ton (3-ton) charcoal adsorber only provides an additional 1.1 percent capacity to the existing offgas system. DCD Table 12.2-16 shows that the annual airborne releases from the offgas system represent only about 4 percent of the total annual airborne releases from Unit 3. Additional charcoal adsorbers would improve the holdup times of the noble gases and C-14, but those only contribute approximately 24 percent to the thyroid dose. Therefore, additional charcoal adsorber material could make a maximum improvement of 0.96 percent of the 28 person-rem/year thyroid dose, or 0.27 person-rem/year. The $9,450/year cost of the 3-ton charcoal adsorber augment divided by the annual dose reduction of 0.27 person-rem/year, results in an estimated cost of over $35,000/person-rem saved. This augment exceeds the cost-benefit ratio of $1000/person-rem prescribed in 10 CFR 50, Appendix I, and is eliminated from further consideration.

Main Condenser Vacuum Pump (MCVP) Charcoal/HEPA Filtration System

DCD Table 12.2-16 shows that the annual airborne iodine releases from the MCVP represent approximately 0.7 percent of the total annual airborne iodine releases from Unit 3. Because the iodines contribute about 73 percent to the 28 person-rem/year thyroid dose, this represents a maximum improvement of 0.5 percent to the thyroid dose, or 0.14 person-rem-year. The $8,170/year cost of the MCVP HEPA filtration system augment divided by the annual dose reduction of 0.14 person-rem/year, results in an estimated cost of over $58,000/person-rem saved. This augment exceeds the cost-benefit ratio of $1000/person-rem prescribed in 10 CFR 50, Appendix I, and is eliminated from further consideration.

600-ft^3 Gas Decay Tank

It is assumed that the gas decay tank is an augment to the offgas system. The flow rate through the offgas system is 54 m^3/hr (31.8 cfm) per DCD Table 12.2-15. As a result, the average residence time in a 600 ft^3 gas decay tank is approximately 19 minutes. While this decay time will have a negligible effect on iodines, particulates, C-14, and H-3, it will mitigate the dose consequences of short-lived noble gases. Because the noble gases contribute 1.5 person-rem/year to the thyroid dose, even complete elimination of the noble gases represents a maximum improvement in the thyroid dose of only 1.5 person-rem/year. The $8,040/year cost of the 600 ft^3 gas decay tank augment divided by the annual dose reduction of 1.5 person-thyroid-rem/year results in an estimated cost of over $5,000/person-thyroid-rem saved. This augment exceeds the cost-benefit ratio of $1000/person-thyroid-rem prescribed in 10 CFR 50, Appendix I, and is eliminated from further consideration.

1000 cfm Charcoal/HEPA Filtration System

As discussed above for 15,000 cfm HEPA filtration systems, the Unit 3 building exhaust system flow rates greatly exceed 472 l/sec (1000 cfm). Therefore, this augment is not effective for Unit 3 and is eliminated from further consideration.

Conclusion

None of the gaseous radwaste augments are cost-beneficial in reducing the annual thyroid dose from gaseous effluents for Unit 3.

11.4 Solid Waste Management System

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

11.4.1 SWMS Design Bases

Replace the seventh bullet of the first paragraph with the following.

NAPS DEP 11.4-1
STD COL 11.4-4-A

- The Radwaste Building provides storage space sized to hold the total combined volume of packaged Class A, B, and C low-level radioactive waste estimated to be generated during six months of plant operations. Such waste is normally promptly disposed of at licensed offsite processing and disposal facilities. In the event that an offsite facility is not available to accept Class B and C waste, the Radwaste Building has been configured to accommodate at least 10 years of packaged Class B and C waste and approximately three months (up to three shipments) of packaged Class A waste, considering routine operations and anticipated operational occurrences. This Class B and C waste storage capacity is based on a conservative estimate of the annual generation of low-level waste, without credit for potential waste minimization techniques and methods other than dewatering. In the event that an offsite facility is not available to accept Class B and C waste, a waste minimization plan will also be implemented. This plan will consider strategies to reduce generation of Class B and C waste, including reducing the in-service run length of resin beds, as well as resin selection, short-loading, and point of generation segregation techniques. Good fuel performance will also reduce fission products in reactor and spent fuel pool water, and hence the volume of Class B and C waste generated. Implementation of these techniques could substantially extend the capacity of the Class B and C storage area in the Radwaste Building. If additional storage capacity for Class B and C waste is required, further temporary storage would be developed in accordance with NUREG-0800, Standard Review Plan 11.4, Appendix 11.4-A.

Add the following after the second paragraph.

STD SUP 11.4-1

The LWMS offsite dose calculations, which are described in Section 12.2.2.4, include the offsite doses from the SWMS liquid effluents, as they are processed by the LWMS. Similarly, the GWMS offsite dose calculations, which are described in Section 12.2.2.2, include the offsite doses from the SWMS gaseous effluents, as they are inputs processed by the GWMS. The cost-benefit analyses in Section 11.2.1 for the LWMS and in Section 11.3.1 for the GWMS address the liquid and gaseous effluents that are generated from solid waste processing by the SWMS. Because these two cost-benefit analyses include the liquid and gaseous effluents from the SWMS, the augments considered for the LWMS and GWMS apply to the SWMS, which provides inputs to those systems. As described in Sections 11.2.1 and 11.3.1, no augments are needed for the LWMS and GWMS to comply with 10 CFR 50, Appendix I, Section II.D. Therefore, no augments are needed for the SWMS to comply with 10 CFR 50, Appendix I, Section II.D.

Replace the fourth sentence of the fourth paragraph with the following:

STD COL 11.4-5-A

Section 12.6 discusses how the ESBWR design features and procedures for operation will minimize contamination of the facility and environment, facilitate decommissioning, and minimize the generation of radioactive wastes, in compliance with 10 CFR 20.1406. Section 13.5 describes the requirement for procedures for operation of the radioactive waste processing system. Operating procedures for mobile/portable SWMS required by Sections 12.4, 12.5, and 13.5 address requirements of 10 CFR 20.1406.

11.4.2.2 System Operation

11.4.2.2.4 Container Storage Subsystem

Replace the first sentence with the following.

NAPS DEP 11.4-1 On-site storage space for packaged waste is provided.

Add the following at the end of this section.

NAPS DEP 11.4-1 On-site storage space for packaged waste is provided in the Radwaste Building. The Radwaste Building waste storage space can accommodate a minimum of ten years of Class B and C waste generated during plant operation, and three months of Class A waste.

The available storage capacity was determined based on anticipated low-level waste volumes generated during plant operation. As a conservative measure, no volume reduction methods or techniques were credited in determination of the volume of Class B and C waste to be stored other than dewatering to meet stabilized waste criteria.

The stored Class B/C HICs are shielded by shield bells surrounding each container and shield wall enclosing the storage area. Shielding analyses, assuming filled HICs and crediting shielding and radioactive decay of the HIC contents over time, have shown that the dose rates in surrounding areas, both within the building and externally, are maintained below the allowable limits in accordance with the radiological area classification as defined in Section 12.3.1.3. Total radioactive material inventory limits are established to ensure shielding analysis assumptions for HIC dose rates are maintained. Inventory records are maintained for waste types, waste contents, radionuclides and radioactive material, dates of storage, shipments, and other relevant data related to storage of Class B and C wastes.

To maintain container integrity for the storage period, and to allow handling during eventual transportation and disposal, the HICs are constructed of corrosion resistant materials that are compatible with the stored waste and the indoor environment of the Radwaste Building. The design life for the HIC is 300 years. HICs are vented to prevent internal pressurization due to gases generated from stored wastes. The vented gases are removed from the storage space by the Radwaste Building heating, ventilating, and air-conditioning system, which is filtered and monitored prior to discharge to atmosphere. Visual inspection is periodically performed for a sampling of HICs using remote monitoring techniques to ensure container integrity in storage.

Class B and C wastes are stored in HICs that meet transportation and disposal requirements in effect at the time the container is placed in storage. In the event that repackaging is required at the time of disposal due to requirements in effect at that time, the HIC can be relocated to a dewatering station for processing.

Fire protection features for the Radwaste Building waste storage area are provided as described in Section 9.5.1, Fire Protection System, and Section 9A, Fire Hazards Analysis. The floor drains in the waste storage area are sized for the fire suppression water anticipated and are directed to the LWMS for processing.

The Class B/C HICs are remotely placed in the storage area utilizing the Radwaste Building crane. Accurate placement and retrieval of the HIC is accomplished using indexing or locating features of the crane. The crane is equipped with a grapple mechanism and load cell for handling the HIC or shield bell.

11.4.2.3 Detailed System Component Description

11.4.2.3.5 SWMS Processing Subsystem

Replace the last three sentences of the second paragraph with the following.

STD COL 11.4-1-A Testing of the SWMS includes testing specified in Table 1 of RG 1.143. Implementation of the programs described in Section 12.1, for maintaining occupational dose ALARA, and Section 12.5, Radiation Protection Program, ensure that operation, maintenance, and testing of the SWMS satisfy the guidance contained in RG 8.8.

STD COL 11.4-2-A Specific equipment connection configuration and plant sampling procedures are used to implement the guidance in Inspection and Enforcement (IE) Bulletin 80-10 (DCD Reference 11.4-19). The permanent and mobile/portable non-radioactive systems, which are connected to radioactive or potentially radioactive portions of SWMS, are protected from contamination with an arrangement of double check valves in each line. The configuration of each line is also equipped with a tell-tale connection, which permits periodic checks to confirm the integrity of the line and its check valve arrangement. Plant procedures describe sampling of non-radioactive systems that could potentially become contaminated by cross-connection with systems that contain radioactive material. In accordance with the guidance in RG 1.109, exposure pathways that may arise due to unique conditions are considered for incorporation into the plant-specific ODCM if they are likely to contribute significantly to the total dose.

STD COL 11.4-3-A Waste classification and process controls are described in the PCP. NEI 07-10, "Generic FSAR Template Guidance for Process Control Program (PCP)," which is under review by the NRC, is incorporated by reference. (Reference 11.4-201) The milestone for development and implementation of the PCP is addressed in Section 13.4.

11.4.6 COL Information

11.4-1-A SWMS Processing Subsystem Regulatory Guide Compliance

STD COL 11.4-1-A This COL item is addressed in Section 11.4.2.3.5.

11.4-2-A Compliance with IE Bulletin 80-10

STD COL 11.4-2-A This COL item is addressed in Section 11.4.2.3.5.

11.4-3-A Process Control Program

STD COL 11.4-3-A This COL item is addressed in Section 11.4.2.3.5.

11.4-4-A Temporary Storage Facility

STD COL 11.4-4-A This COL item is addressed in Section 11.4.1.

11.4-5-A Compliance with Part 20.1406

STD COL 11.4-5-A This COL item is addressed in Section 11.4.1.

11.4.7 References

11.4-201 NEI 07-10, Generic FSAR Template Guidance for Process Control Program (PCP).

11.5 Process Radiation Monitoring System

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

Add the following paragraph at the end of this section.

STD COL 11.5-3-A Replace text references to DCD Table 11.5-5 with Table 11.5-201.

11.5.4.4 Setpoints

Replace the first sentence in this section with the following.

STD COL 11.5-2-A The derivation of setpoints used for offsite dose monitors are described in the ODCM. Refer to Section 11.5.4.5 for a discussion regarding ODCM development and implementation.

11.5.4.5 Offsite Dose Calculation Manual

Replace this section with the following.

STD COL 11.5-2-A The methodology and parameters used for calculation of offsite dose and monitoring are described in the ODCM. NEI 07-09, Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description, which is under review by the NRC, is incorporated by reference. (Reference 11.5-201) The milestone for development and implementation of the ODCM is addressed in Section 13.4. The provisions for sampling liquid and gaseous waste streams identified in Table 11.5-201 and DCD Table 11.5-6 will be included in the ODCM.

11.5.4.6 Process and Effluent Monitoring Program

Replace this section with the following.

STD COL 11.5-3-A The program for process and effluent monitoring and sampling is described in the ODCM. Refer to Section 11.5.4.5 for a discussion regarding ODCM development and implementation.

11.5.4.7 Sensitivity or Subsystem Lower Limit of Detection

Replace this section with the following.

STD COL 11.5-1-A The ODCM describes the methodology for deriving the lower limit of detection for each effluent monitor. Refer to Section 11.5.4.5 for a discussion regarding ODCM development and implementation. The estimated sensitivities (i.e., the dynamic detection ranges) of process radiation monitors are described in DCD Tables 11.5-2 and 11.5-4. The bases for these values are provided in DCD Table 11.5-9. These ranges are adjusted according to unique plant configurations and radiation background in accordance with written procedures. The processes described in these procedures are consistent with the bases defined in DCD Table 11.5-9. If changes to the values in DCD Tables 11.5-2 or 11.5-4 are necessary, the FSAR is updated to reflect these new values.

11.5.4.8 Site Specific Offsite Dose Calculation

Replace this section with the following.

STD COL 11.5-4-A

10 CFR 50, Appendix I guidelines are addressed in the ODCM. Refer to Section 11.5.4.5 for a discussion regarding ODCM development and implementation.

Site-specific evaluations for dose to members of the public are addressed in Section 12.2.

11.5.4.9 Instrument Sensitivities

Replace this section with the following.

STD COL 11.5-5-A

The sensitivities, sampling and analytical frequencies and bases for each gaseous and liquid sample are described in the ODCM. Refer to Section 11.5.4.5 for a discussion regarding ODCM development and implementation.

11.5.5.8 Setpoints

Replace this section with the following:

STD COL 11.5-2-A

Refer to Section 11.5.4.4.

Replace DCD Table 11.5-5 with Table 11.5-201.

11.5.7 COL Information

11.5-1-A Sensitivity or Subsystem Lower Limit of Detection

STD COL 11.5-1-A

This COL item is addressed in Section 11.5.4.7.

11.5-2-A Offsite Dose Calculation Manual

STD COL 11.5-2-A This COL item is addressed in Sections 11.5.4.4, 11.5.4.5, and 11.5.5.8.

11.5-3-A Process and Effluent Monitoring Program

STD COL 11.5-3-A This COL item is addressed in Sections 11.5 and 11.5.4.6, and Table 11.5-201.

11.5-4-A Site Specific Offsite Dose Calculation

STD COL 11.5-4-A This COL item is addressed in Section 11.5.4.8.

11.5-5-A Instrument Sensitivities

STD COL 11.5-5-A This COL item is addressed in Section 11.5.4.9.

11.5.8 References

11.5-201 NEI 07-09, “Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description”

DCD Table 11.5-2

Replace the ** note with the following.

STD COL 11.5-3-A Activity levels are expected to be at the subsystem’s lower limit of detection (LLD). Applicable values are included in the plant-specific ODCM. See Section 12.2 for expected activity of various processes and effluents.

DCD Table 11.5-4

Replace the ** note with the following.

STD COL 11.5-3-A Activity levels are expected to be at the subsystem’s LLD. Applicable values are included in the plant-specific ODCM. See Section 12.2 for expected activity of various processes and effluents.

STD COL 11.5-3-A

Table 11.5-201 Provisions for Sampling Liquid Streams

No.	Process Systems as listed in NUREG-0800, SRP 11.5 Table 2 (Draft Rev. 4)	ESBWR System (s) that Perform the Equivalent SRP 11.5 Function (Note 1)	In Process	In Effluent	
			Grab Notes 2 & 7	Grab Notes 2 & 7	Continuous Notes 2 & 7
1.	Liquid Radwaste (Batch) Effluent System Note 3	Equipment (Low Conductivity Drain Subsystem Floor (High Conductivity) Drain Subsystem Detergent Drain Subsystem	S&A	S&A, H3 Note 4	-
2.	Service Water System and/or Circulating Water System	Plant Service Water System and Circulating Water System	-	S&A, H3 Note 9	-
3.	Component Cooling Water System	Reactor Component Cooling Water System	S&A	S&A H3	(S&A) Notes 6 & 8
4.	Spent Fuel Pool Treatment System	Spent Fuel Pool Treatment System	S&A	S&A H3	(S&A) Notes 6 & 8
5.	Equipment & Floor Drain Collection and Treatment Systems	LCW Drain Subsystem HCW Drain Subsystem Detergent Drain Subsystem Chemical Waste Drain Subsystem Reactor Component Cooling Water System (RCCWS) Drain Subsystem	-	S&A H3	(S&A) Notes 6 & 8
6.	Phase Separator Decant & Holding Basin Systems	Equipment (Low Conductivity) Drain Subsystem Floor (High) Drain Subsystem	-	S&A H3	(S&A) Notes 6 & 8
7.	Chemical & Regeneration Solution Waste Systems	Chemical Waste Drain Subsystem	-	S&A H3	(S&A) Notes 6 & 8
8.	Laboratory & Sample System Waste Systems	Chemical Waste Drain Subsystem	-	S&A H3	(S&A) Notes 6 & 8

STD COL 11.5-3-A

Table 11.5-201 Provisions for Sampling Liquid Streams

No.	Process Systems as listed in NUREG-0800, SRP 11.5 Table 2 (Draft Rev. 4)	ESBWR System (s) that Perform the Equivalent SRP 11.5 Function (Note 1)	In Process	In Effluent	
			Grab Notes 2 & 7	Grab Notes 2 & 7	Continuous Notes 2 & 7
9.	Laundry & Decontamination Waste Systems	Detergent Drain Subsystem	-	S&A H3	(S&A) Notes 6 & 8
10.	Resin Slurry, Solidification & Baling Drain Systems	Equipment (Low Conductivity) Drain Subsystem, Floor (High) Drain Subsystem	-	S&A H3	(S&A) Notes 6 & 8
11.	Storm & Underdrain Water System	Storm Drains	-	S&A, H3 Notes 3 & 10	-
12.	Tanks and Sumps Inside Reactor Building	Equipment (Low Conductivity) Drain Subsystem Floor (High) Drain Subsystem Chemical Waste Drain Subsystem Detergent Drain Subsystem	-	S&A H3	(S&A) Notes 6 & 8
13.	Ultrasonic Resin Cleanup Waste Systems	Note 5	-	Note 5	Note 5
14.	Non-Contaminated Waste Water System	Sanitary Waste Discharge System	-	S&A, H3 Note 11	-
15.	Liquid Radioactive Waste Processing Systems (Includes Reverse Osmosis Systems)	Liquid Radioactive Waste Processing Systems (Includes Reverse Osmosis Systems)	S&A	(S&A, H3)	(S&A) Notes 6 & 8

STD COL 11.5-3-A

Table 11.5-201 Provisions for Sampling Liquid Streams

Notes for Table 11.5-201:

1. Table 11.5-201 addresses sampling provisions for ESBWRs, as recommended in Table 2 of SRP 11.5 for BWRs. For process systems identified for BWRs in SRP 11.5 Table 2, but not shown in Table 11.5-201, those systems are not applicable to ESBWR. In some cases, there are multiple subsystems that are used to perform the overall equivalent SRP function and are listed as such in the column.
2. S&A = Sampling & Analysis of radionuclides, to include gross radioactivity, identification and concentration of principal radionuclides and concentration of alpha emitters; R = Gross radioactivity (beta radiation, or total beta plus gamma); H3 = Tritium
3. Liquid Radwaste is processed on a batch-wise basis. The Liquid Waste Management System sample tanks can be sampled for analysis of the batch. See DCD Section 11.2.2.2 for more information on Liquid Radwaste Management.
4. Monitoring of effluents from the Equipment, Floor, and Detergent Drain Subsystems is included in the Offsite Dose Calculation Manual.
5. The ESBWR does not include ultrasonic resin cleanup waste system at this time. Should one be installed, the Liquid Waste Management System would provide sampling and monitoring provisions.
6. The use of parenthesis indicates that these provisions are required only for the systems not monitored, sampled, or analyzed (as indicated) prior to release by downstream provisions.
7. The sensitivity of detection, also defined here as the Lower Limit of Detection (LLD), for each indicated measured variable, is based on the applicable radionuclide (or collection of radionuclides as applicable) as given in ANSI/IEEE N42.18.
8. Processed through radwaste Liquid Waste Management System (LWMS) prior to discharge. Therefore, this process system is monitored, sampled, or analyzed prior to release by downstream provisions. See Note 6 above. Depending on Utility's discretion, additional sampling lines may be installed. Continuous Effluent sampling is not required per Standard Review Plan 11.5 Draft Rev. 4, April 1996, Table 2 for this system function.
9. Grab samples can be obtained from a cooling tower basin. See Section 9.2.1.2 for the PSWS cooling tower basin and Section 10.4.5.2.3 for the Circulating Water System cooling tower basin.
10. Grab samples can be obtained from the Condensate Storage Tank (CST) basin sump. See DCD Section 9.2.6.2.
11. Grab samples can be obtained from the sewage treatment plant. See Section 9.2.4.2.

Chapter 12 Radiation Protection

12.1 Ensuring That Occupational Radiation Exposures Are ALARA

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

Add the following at the beginning of this section.

STD SUP 12.1-1 The ALARA program is addressed in Appendices 12AA and 12BB.

12.1.1.3.1 Compliance with Regulatory Guide 8.8

Replace the first paragraph of this section with the following.

STD COL 12.1-4-A Compliance with Regulatory Guide 8.8 is addressed in Appendix 12BB.

12.1.1.3.2 Compliance with Regulatory Guide 8.10

Replace this section with the following.

STD COL 12.1-1-A Compliance with Regulatory Guide 8.10 is addressed in Appendix 12BB.

12.1.1.3.3 Compliance with Regulatory Guide 1.8

Replace this section with the following.

STD COL 12.1-2-A Compliance with Regulatory Guide 1.8 is addressed in Appendix 12BB.

12.1.3 Operational Considerations

Replace this section with the following.

STD COL 12.1-3-A ALARA program implementation is addressed in Appendix 12BB.

12.1.4 COL Information

12.1-1-A Regulatory Guide 8.10

STD COL 12.1-1-A This COL item is addressed in Section 12.1.1.3.2 and Appendix 12BB.

12.1-2-A Regulatory Guide 1.8

STD COL 12.1-2-A This COL item is addressed in Section 12.1.1.3.3 and Appendix 12BB.

12.1-3-A **Operational Considerations**

STD COL 12.1-3-A This COL item is addressed in Section 12.1.3 and Appendix 12BB.

12.1-4-A **Regulatory Guide 8.8**

STD COL 12.1-4-A This COL item is addressed in Section 12.1.1.3.1 and Appendix 12BB.

12.2 Plant Sources

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

12.2.1.5 Other Contained Sources

Replace this section with the following.

STD COL 12.2-4-A In addition to the contained sources identified above, additional contained sources which contain by-product, source, or special nuclear materials may be maintained on site. These contained sources are used as calibration, check, or radiography sources. These sources are not part of the permanent plant design, and their control and use are governed by plant procedures. The procedures consider the guidance provided in RG 8.8 to ensure that occupational doses from the control and use of the sources are as low as is reasonably achievable (ALARA).

Various types and quantities of radioactive sources are employed to calibrate the process and effluent radiation monitors, the area radiation monitors, and portable and laboratory radiation detectors. Check sources that are integral to the area, process, and effluent monitors consist of small quantities of by-product material and do not require special handling, storage, or use procedures for radiation protection purposes. The same consideration applies to solid and liquid radionuclide sources of exempt quantities or concentrations which are used to calibrate or check the portable and laboratory radiation measurement instruments.

Instrument calibrators are normally used for calibrating gamma dose rate instrumentation. These may be self-contained, heavily shielded, multiple source calibrators. Beta and alpha radiation sources are also available for instrument calibration. Calibration sources are traceable to the National Institute of Standards and Technology, or equivalent.

Radiography sources are surveyed upon entry to the site. Radiation protection personnel maintain copies of the most recent leak test records for owner-controlled sources. Contractor radiography personnel provide copies of the most recent leak test records upon radiation protection personnel request. Radiography is conducted in accordance with approved procedures.

12.2.2.1 Airborne Releases Offsite

Replace this section with the following.

NAPS COL 12.2-2-A

Design basis noble gas, iodine, and other fission product concentrations are taken from the tables in Chapter 11. Airborne sources for normal operating releases are calculated using the source terms given in DCD Section 11.1.

The bases for the airborne sources calculations are provided in Table 12.2-15R. The bases include values used in calculating the annual airborne release source terms provided in DCD Table 12.2-16. The methodology of NUREG-0016 was used in determining the annual airborne release values presented in DCD Table 12.2-16.

Annual Releases

Based on the inputs and criteria described above, the annual airborne releases for Unit 3 normal operations and the Unit 3 airborne concentrations at the site boundary are provided in Table 12.2-17R. This table also shows the maximum activity concentration for each nuclide at the site boundary from the combined operation of Units 1, 2, and 3, and the corresponding concentration limit for the NAPS site from 10 CFR 20, Appendix B, Table 2, Column 1.

12.2.2.2 Airborne Dose Evaluation Offsite

Replace this section with the following.

NAPS COL 12.2-2-A

The bases for the calculation of Unit 3-specific airborne offsite doses are provided in Table 12.2-18aR. The annual gaseous pathway doses are provided in Table 12.2-18bR. The methodology of RG 1.109 was used in determining the annual airborne dose values. The bases include values that are default parameters in RG 1.109 and other values that are Unit 3 site-specific inputs.

The results of the Unit 3 gaseous pathway dose analysis are given in Table 12.2-18bR.

12.2.2.2.1 Compliance with 10 CFR 50, Appendix I, Sections II.B and II.C

Table 12.2-201 demonstrates that offsite doses due to Unit 3 radioactive airborne effluents comply with the regulatory dose limits in 10 CFR 50, Appendix I, Sections II.B and II.C.

NAPS ESP COL 11.1-1

12.2.2.2.2 Compliance with 10 CFR 50, Appendix I, Section II.D

Population dose is determined for the gaseous effluent releases from Unit 3 for both total body dose and thyroid dose. The total body dose is 7.7 person-rem/yr as shown in Table 12.2-204. The thyroid dose is 28 person-rem/yr. The cost-benefit analysis performed to consider gaseous radwaste augments to reduce doses due to gaseous effluents is presented in Section 11.3. Based on the results from the cost-benefit analysis, no augments are cost-beneficial. Therefore, Unit 3 complies with 10 CFR 50, Appendix I, Section II.D.

12.2.2.2.3 Compliance with 10 CFR 20, Appendix B, Table 2, Column 1

Table 12.2-17R provides the gaseous effluent concentrations in comparison to the 10 CFR 20, Appendix B, Table 2, Column 1 limits. The Unit 3 gaseous effluent concentrations comply with 10 CFR 20, Appendix B, Table 2, Column 1.

12.2.2.2.4 Compliance with 10 CFR 20.1301 and 20.1302

Compliance with 10 CFR 20.1301 and 20.1302 is demonstrated in Sections 12.2.2.4.4 and 12.2.2.4.5, respectively.

NAPS ESP COL 11.1-1

12.2.2.2.5 Comparison of ESP Application to Unit 3 Gaseous Effluent Concentrations

As described in Section 12.2.2.1, the radioactive gaseous effluent concentrations for Unit 3 are provided in Table 12.2-17R.

The radioactive gaseous effluent concentrations for the ESPA are included in ESP-ER Table 5.4-7. That table presents the composite annual release activities and activity concentrations of gaseous effluents for a single unit, but is based on a composite of possible radionuclide releases from many reactor designs. The values in that table are the maximum annual activity and corresponding concentration for each radionuclide from the many reactor designs considered.

While ESP-ER Table 5.4-7 contains more radionuclides than Table 12.2-17R due to the use of the composite set of nuclides, the calculated radioactive gaseous effluent concentration for each Unit 3 radionuclide is bounded by the concentration for that nuclide in the ESP-ER. Not only is each radionuclide bounded, the total gaseous effluent release activity for Unit 3 is much less than the total composite release activity considered in the ESP-ER.

12.2.2.2.6 Comparison of ESPA to Unit 3 Gaseous Effluent Doses

As described in Section 12.2.2.2, the calculated radioactive gaseous effluent doses for Unit 3 are provided in Table 12.2-18bR.

The radioactive gaseous effluent doses for the ESP Application are included in ESP-ER Table 5.4-9. The results from that table are reproduced in Table 12.2-18bR.

For both the composite releases used in the ESP-ER, and the Unit 3 normal operating releases, Table 12.2-18bR presents doses to the maximally exposed adult, teenager, child, and infant for the following pathways:

- Nearest site boundary
- Nearest vegetable garden
- Nearest residence
- Nearest meat cow

For the milk pathway, no milk animals are within 8 km (5 miles) of Unit 3.

As noted in Section 2.3.5, the distance to the site boundary has been measured using GIS and although it is known to be farther than the value used in the ESP-ER, the ESP-ER value is conservatively used in calculating Unit 3 gaseous effluent doses at the site boundary.

The locations of the nearest vegetable garden, residence, and meat cow were updated since the ESP-ER and closer locations than addressed in the ESP-ER were identified. For these pathways, the closest location from all three of the pathways was used for the distance to the MEI for each pathway.

While the total activity in the gaseous radioactive effluents for Unit 3 is much less than that estimated in the ESP-ER, the calculated doses for some of the pathways shown in Table 12.2-18bR are not lower due to the reductions in the distances to the MEI receptor locations as described above. Values in Table 12.2-18bR in bold print indicate pathways for

NAPS ESP VAR 12.2-1

which the estimated Unit 3 ESBWR dose to the MEI is larger than the corresponding ESP-ER composite release dose to the MEI.

Although some pathways in Table 12.2-18bR show slight increases in total body and thyroid doses to the MEI from the changes in MEI locations, Table 12.2-18bR summarizes the annual total body, thyroid, and skin doses to the MEI for the garden, residence, and meat cow pathways, and Table 12.2-201 shows that the Unit 3 doses are lower than those calculated and presented in ESP-ER Table 5.4-10.

12.2.2.4 Liquid Doses Offsite

Replace this section with the following.

NAPS COL 12.2-3-A

Liquid pathway doses were calculated based on the criteria specified in DCD Section 12.2.2.3 for compliance with 10 CFR 50, Appendix I. Dose conversion factors and methodologies consistent with RGs 1.109 and 1.113 were used as described in DCD References 12.2-7 and 12.2-4, respectively.

The liquid effluent pathway offsite dose calculation bases are provided in Table 12.2-20aR. The bases include values that are default parameters in RG 1.109 and other values that are Unit 3 site-specific inputs.

Based on the annual liquid release offsite values in DCD Table 12.2-19b, which are repeated in Table 12.2-19bR, the Unit 3 annual liquid release concentrations were calculated based upon the criteria specified in DCD Section 12.2.2.3 and the Unit 3-specific input values shown in Table 12.2-20aR. Table 12.2-19bR also shows the maximum activity concentration for each nuclide at the end of the discharge canal from the combined operation of Units 1, 2, and 3, and the corresponding concentration limit for the NAPS site from 10 CFR 20, Appendix B, Table 2, Column 2.

The LADTAPII code is used to perform the liquid effluent dose analysis (DCD Reference 12.2-3). The results of the dose calculation are given in Table 12.2-20bR.

12.2.2.4.1 Compliance with 10 CFR 50, Appendix I, Section II.A

Table 12.2-202 demonstrates that offsite doses due to Unit 3 radioactive liquid effluents comply with the regulatory dose limits in 10 CFR 50, Appendix I, Section II.A.

NAPS ESP COL 11.1-1

12.2.2.4.2 Compliance with 10 CFR 50, Appendix I, Section II.D

Population dose is determined for the liquid effluent releases from Unit 3 for both total body dose and thyroid dose. The total body dose is 1.0 person-rem/yr as shown in Table 12.2-204. The thyroid dose is 0.69 person-rem/yr. The cost-benefit analysis performed to consider liquid radwaste augments to reduce doses due to liquid effluents is presented in Section 11.2. Based on the above liquid effluent dose estimate values and the threshold value from the cost-benefit analysis, no augments are cost-beneficial. Therefore, Unit 3 complies with 10 CFR 50, Appendix I, Section II.D.

12.2.2.4.3 Compliance with 10 CFR 20, Appendix B, Table 2, Column 2

Compliance with 10 CFR 20, Appendix B, Table 2, Column 2 is demonstrated in Table 12.2-19bR.

12.2.2.4.4 Compliance with 10 CFR 20.1301 and 20.1302

This section demonstrates that offsite doses due to Unit 3, combined with offsite doses due to Units 1 and 2 and the NAPS independent spent fuel storage installation (ISFSI), comply with the regulatory limits in 10 CFR 20.1301 for doses to members of the public.

Using the Unit 3-specific gaseous effluent release activities identified in Table 12.2-17R, and the Unit 3-specific liquid effluent release activities identified in Table 12.2-19bR, the total annual doses to the MEI and the population resulting from Unit 3 liquid and gaseous effluents are calculated and presented in Tables 12.2-203 and 12.2-204, respectively.

The direct radiation contribution from operation of Unit 3 is negligible. The direct dose contribution from Unit 3 at two distances is provided in DCD Table 12.2-21. That table shows the annual dose at 1000 m (0.62 mi) to be 1.66E-06 mSv/yr (1.66E-04 mrem/yr). Section 9.3.9 shows that Unit 3 uses hydrogen water chemistry, and DCD Section 12.2.1.3 explains that the direct dose contribution takes into account hydrogen water chemistry. The distance from Unit 3 to the nearest residence is 1191 m (0.74 mi) in the NW direction, as shown in Table 2.3-15R. The distance from Unit 3 to the location on the site boundary with the highest gaseous effluent annual dose is 1416 m (0.88 mile) in the ESE direction. This is the distance from Unit 3 to the site boundary, that is, the exclusion area boundary (EAB) in the direction of maximum annual χ/Q, as shown in Table 2.3-16R. These distances from Unit 3 to each type of receptor location are greater than those presented in the DCD, so the Unit 3 direct radiation dose rate at each location us even lower than the very low rate cited above for 1000 m (0.62 mi).

The total annual doses to the MEI resulting from North Anna Units 1 and 2 liquid and gaseous effluents are provided in Table 12.2-203. The values shown are representative based on review of Units 1 and 2 annual radiological environmental operating reports (e.g., Reference 12.2-203).

The direct radiation contribution from operation of Units 1 and 2 is negligible. An evaluation of operating plants by the NRC states that:

> "…because the primary coolant of an LWR is contained in a heavily shielded area, dose rates in the vicinity of light water reactors are generally undetectable and are less than 1 mrem/year at the site boundary."

The NRC concludes that the direct radiation from normal operation results in "small contributions at site boundaries" (Reference 12.2-204, Section 4.6.1.2). For the NAPS site, the nearest residence is at a distance typical of a site boundary evaluated by NRC. An assumed value of 1 mrem/yr is included in Table 12.2-203 to account for the dose to the MEI at the nearest residence from operation of Units 1 and 2.

Discharged fuel assemblies from NAPS Units 1 and 2 are stored in the NAPS ISFSI (Reference 12.2-205). The direct radiation contribution from operation of the NAPS ISFSI is small, both at the residence nearest to the ISFSI, which is south and slightly east of the ISFSI at about 870 m (0.54 mi), and at the closest point to the site boundary, which is south and slightly west of the ISFSI at approximately 760 m (0.47 mi). The annual contribution at the site boundary from the ISFSI is no more than 3.6E-02 mSv/yr (3.6 mrem/yr). This value is based on a conservatively estimated peak dose rate from a fully-filled ISFSI with 84 casks/modules containing NAPS Units 1 and 2 fuel assemblies and the distance from the ISFSI to the site boundary, which is shorter than that to the residence nearest the ISFSI. This ISFSI dose contribution is then conservatively applied to the MEI for the nearest residence from Unit 3, which is 1191 m (0.74 mi) in the NW direction and even further from the ISFSI.

Table 12.2-203 shows that the total NAPS site doses resulting from the normal operation of Units 1, 2, and 3 and applied at the nearest residence are well within the regulatory limits of 40 CFR 190. These doses are applied at the distance to the nearest residence from Unit 3, that is, 1191 m (0.74 mi), but in the direction of the maximum annual χ/Q, that is, in the ESE direction, and using the maximum D/Q, which is from the NNE direction. These doses bound those at the site boundary.

Table 12.2-204 shows the total body doses from liquid and gaseous effluents doses attributable to Unit 3 for the population within 50 miles of the NAPS site.

12.2.2.4.5 **Compliance with 10 CFR 20.1302**

Surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas are conducted to demonstrate compliance with the dose limits given in 10 CFR 20.1302 for individual members of the public.

Compliance with the annual dose limit in 10 CFR 20.1302 is demonstrated by showing that the calculated total effective dose equivalent to the individual likely to receive the highest dose does not exceed the annual dose limit.

NAPS ESP COL 11.1-1

12.2.2.4.6 **Comparison of ESPA to NAPS Site with Unit 3 Liquid Effluent Concentrations**

As described in Section 12.2.2.4, the radioactive liquid effluent concentrations for Unit 3 are provided in Table 12.2-19bR. This table also shows the maximum activity concentration for each nuclide at the end of the discharge canal from the combined operation of Units 1, 2, and 3, and the corresponding concentration limit for the NAPS site.

The radioactive liquid effluent concentrations for the NAPS site from the combined operation of the two new units and the existing units as presented in the ESPA are included in ESP-ER Table 5.4-6. That table presents the composite annual release activities of liquid effluents for a single new unit, but based on a composite of possible radionuclide releases from many reactor designs. For all isotopes except tritium, the maximum annual activity for each radionuclide is the maximum from the many different types of reactor designs considered. ESP-ER Table 5.4-6 contains more radionuclides than Table 12.2-19bR due to the use of the composite set of nuclides in the ESP-ER.

NAPS ESP VAR 12.2-3

For most radionuclides in the Unit 3 liquid effluent, the maximum activity is bounded by the activity for that nuclide in the ESP-ER. Annual release activities in bold print in Table 12.2-19bR indicate those 12 radionuclides for which the estimated Unit 3 release activity is slightly greater than the composite release activity as presented in the ESP-ER.

Although not every radionuclide is bounded, the total liquid effluent release activity of Unit 3 is less than the total composite release activity presented in the ESP-ER.

Table 12.2-19bR shows the total activity concentrations at the site release point for the nuclides in radioactive liquid effluent for Units 1, 2, and 3. For every nuclide, the maximum activity concentration is equal to or less than the corresponding value in ESP-ER Table 5.4-6.

12.2.2.4.7 Comparison of ESPA to Unit 3 Liquid Effluent Doses

As described in Section 12.2.2.4, the calculated radioactive liquid effluent doses for Unit 3 are provided in Table 12.2-20bR.

The radioactive liquid effluent doses for the ESPA are included in ESP-ER Table 5.4-8. The results from that table are reproduced in Table 12.2-20bR. The dose for each liquid radioactive effluent pathway for Unit 3 is less than the corresponding estimate in the ESP-ER. Table 12.2-202 summarizes the annual total body and bone doses to the MEI and shows that the Unit 3 doses are lower than those calculated and presented in ESP-ER Table 5.4-10.

As indicated in Tables 12.2-203 and 12.2-204, the annual total site doses to the MEI and the population within 50 miles of Unit 3 are lower than those calculated and presented in ESP-ER.

12.2.4 COL Information

12.2-2-A Airborne Effluents and Doses

NAPS COL 12.2-2-A

This COL item is addressed in Sections 12.2.2.1, 12.2.2.2, and Table 2.0-201.

12.2-3-A Liquid Effluents and Doses

NAPS COL 12.2-3-A

This COL item is addressed in Section 12.2.2.4.

12.2-4-A Other Contained Sources

STD COL 12.2-4-A

This COL item is addressed in Section 12.2.1.5.

12.2.5 References

12.2-201 [Deleted]

12.2-202 [Deleted]

12.2-203 Virginia Electric and Power Company, North Anna Units 1 & 2 and Independent Spent Fuel Storage Installation (ISFSI) Annual Radiological Environmental Operating Report, April 17, 2006.

12.2-204 NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, U. S. Nuclear Regulatory Commission, May 1996.

12.2-205 Virginia Electric and Power Company, North Anna Independent Spent Fuel Storage Installation, Final Safety Analysis Report, Revision 6, Docket No. 72-16, License No. 2507, June 2008.

NAPS COL 12.2-2-A
NAPS ESP COL 11.1-1

Table 12.2-15R Airborne Sources Calculation

Calculation Bases

Methodology	**DCD Appendix 12B**
Noble Gas Source at t=30 min	740 MBq/sec (20,000 μCi/sec)
I^{131} Release Rate	3.7 MBq/sec (100 μCi/sec)
Directions and distances from site to receptor locations	See Table 2.3-16R
Meteorology χ/Q	See Table 2.3-16R
Meteorology D/Q	See Table 2.3-16R
Plant Availability Factor	0.92
Offgas System	
Offgas stream temperature	100°F
Flow rate at 100°F	54 m^3/hr
K_d (Kr)	18.5 cm^3/g
K_d (Xe)	330 cm^3/g
K_d (Ar)	6.4 cm^3/g
Guard tank charcoal mass	7,500 kg (single tank)
Adsorber tank charcoal mass	27,750 kg (each)
Adsorber tank arrangement	2 parallel trains of 4 tanks each
Turbine Gland Sealing System Exhaust	
I-131 release	0.81 Ci/yr per μCi/g of I-131 in coolant
I-133 release	0.22 Ci/yr per μCi/g of I-133 in coolant

NAPS COL 12.2-2-A
NAPS ESP COL 11.1-1

Table 12.2-17R Comparison of Airborne Release Concentrations with 10 CFR 20 Limit

Nuclide	Unit 3 Annual Release		Unit 3 Concentration		Units 1, 2 & 3 Concentration	ECL	Units 1, 2, & 3 Fraction of ECL
	MBq/yr	Ci/yr	Bq/m^3	µCi/ml	µCi/ml	µCi/ml	
Kr-83m	8.5E+01	2.3E-03	1.0E-05	2.7E-16	2.7E-16	5.0E-05	5.4E-12
Kr-85m	6.6E+05	1.8E+01	7.7E-02	2.1E-12	7.2E-11	1.0E-07	7.2E-04
Kr-85	5.2E+06	1.4E+02	6.1E-01	1.6E-11	1.3E-09	7.0E-07	1.8E-03
Kr-87	1.4E+06	3.8E+01	1.6E-01	4.4E-12	4.4E-11	2.0E-08	2.2E-03
Kr-88	2.1E+06	5.7E+01	2.5E-01	6.7E-12	1.3E-10	9.0E-09	1.5E-02
Kr-89	1.4E+07	3.8E+02	1.6E+00	4.4E-11	4.4E-11	1.0E-09	4.4E-02
Xe-131m	1.5E+05	4.1E+00	1.8E-02	4.8E-13	2.3E-12	2.0E-06	1.1E-06
Xe-133m	1.9E+02	5.1E-03	2.2E-05	6.0E-16	1.0E-10	6.0E-07	1.7E-04
Xe-133	4.1E+07	1.1E+03	4.8E+00	1.3E-10	9.3E-09	5.0E-07	1.9E-02
Xe-135m	2.2E+07	5.9E+02	2.6E+00	7.0E-11	7.7E-11	4.0E-08	1.9E-03
Xe-135	2.8E+07	7.6E+02	3.3E+00	8.9E-11	3.0E-10	7.0E-08	4.3E-03
Xe-137	2.8E+07	7.6E+02	3.3E+00	8.9E-11	8.9E-11	1.0E-09	8.9E-02
Xe-138	2.3E+07	6.2E+02	2.7E+00	7.3E-11	9.5E-11	2.0E-08	4.7E-03
I-131	8.4E+03	2.3E-01	9.9E-04	2.7E-14	2.6E-13	2.0E-10	1.3E-03
I-132	5.8E+04	1.6E+00	6.8E-03	1.8E-13	2.3E-13	2.0E-08	1.1E-05
I-133	4.2E+04	1.1E+00	4.9E-03	1.3E-13	4.2E-13	1.0E-09	4.2E-04
I-134	1.1E+05	3.0E+00	1.3E-02	3.5E-13	3.7E-13	6.0E-08	6.1E-06
I-135	5.9E+04	1.6E+00	6.9E-03	1.9E-13	3.0E-13	6.0E-09	5.0E-05

NAPS COL 12.2-2-A
NAPS ESP COL 11.1-1

Table 12.2-17R Comparison of Airborne Release Concentrations with 10 CFR 20 Limit

	Unit 3 Annual Release		Unit 3 Concentration		Units 1, 2 & 3 Concentration	ECL	Units 1, 2, & 3 Fraction of ECL
Nuclide	MBq/yr	Ci/yr	Bq/m^3	µCi/ml	µCi/ml	µCi/ml	
H-3	2.8E+06	7.6E+01	3.3E-01	8.9E-12	8.9E-12	1.0E-07	8.9E-05
C-14	5.3E+05	1.4E+01	6.2E-02	1.7E-12	1.7E-12	3.0E-09	5.6E-04
Na-24	5.4E+00	1.5E-04	6.3E-07	1.7E-17	1.7E-17	7.0E-09	2.4E-09
P-32	1.3E+00	3.5E-05	1.5E-07	4.1E-18	4.1E-18	5.0E-10	8.2E-09
Ar-41	1.4E+03	3.8E-02	1.6E-04	4.4E-15	4.4E-15	1.0E-08	4.4E-07
Cr-51	1.8E+02	4.9E-03	2.1E-05	5.7E-16	5.7E-16	3.0E-08	1.9E-08
Mn-54	1.5E+02	4.1E-03	1.8E-05	4.8E-16	4.8E-16	1.0E-09	4.8E-07
Mn-56	1.1E+01	3.0E-04	1.3E-06	3.5E-17	3.5E-17	2.0E-08	1.7E-09
Fe-55	4.7E+01	1.3E-03	5.5E-06	1.5E-16	1.5E-16	3.0E-09	5.0E-08
Fe-59	2.0E+01	5.4E-04	2.3E-06	6.3E-17	6.3E-17	5.0E-10	1.3E-07
Co-58	4.0E+01	1.1E-03	4.7E-06	1.3E-16	1.3E-16	1.0E-09	1.3E-07
Co-60	3.2E+02	8.6E-03	3.8E-05	1.0E-15	1.0E-15	5.0E-11	2.0E-05
Ni-63	4.7E-02	1.3E-06	5.5E-09	1.5E-19	1.5E-19	1.0E-09	1.5E-10
Cu-64	6.9E+00	1.9E-04	8.1E-07	2.2E-17	2.2E-17	3.0E-08	7.3E-10
Zn-65	3.2E+02	8.6E-03	3.8E-05	1.0E-15	1.0E-15	4.0E-10	2.5E-06
Rb-89	2.0E-01	5.4E-06	2.3E-08	6.3E-19	6.3E-19	2.0E-07	3.2E-12
Sr-89	1.5E+02	4.1E-03	1.8E-05	4.8E-16	4.8E-16	2.0E-10	2.4E-06
Sr-90	1.0E+00	2.7E-05	1.2E-07	3.2E-18	3.2E-18	6.0E-12	5.3E-07

NAPS COL 12.2-2-A
NAPS ESP COL 11.1-1

Table 12.2-17R Comparison of Airborne Release Concentrations with 10 CFR 20 Limit

	Unit 3 Annual Release		Unit 3 Concentration		Units 1, 2 & 3 Concentration	ECL	Units 1, 2, & 3 Fraction of ECL
Nuclide	MBq/yr	Ci/yr	Bq/m^3	µCi/ml	µCi/ml	µCi/ml	
Y-90	8.1E-02	2.2E-06	9.5E-09	2.6E-19	2.6E-19	9.0E-10	2.9E-10
Sr-91	6.7E+00	1.8E-04	7.9E-07	2.1E-17	2.1E-17	5.0E-09	4.2E-09
Sr-92	4.6E+00	1.2E-04	5.4E-07	1.5E-17	1.5E-17	9.0E-09	1.6E-09
Y-91	1.7E+00	4.6E-05	2.0E-07	5.4E-18	5.4E-18	2.0E-10	2.7E-08
Y-92	3.7E+00	1.0E-04	4.3E-07	1.2E-17	1.2E-17	1.0E-08	1.2E-09
Y-93	7.2E+00	1.9E-04	8.4E-07	2.3E-17	2.3E-17	3.0E-09	7.6E-09
Zr-95	4.4E+01	1.2E-03	5.2E-06	1.4E-16	1.4E-16	4.0E-10	3.5E-07
Nb-95	2.4E+02	6.5E-03	2.8E-05	7.6E-16	7.6E-16	2.0E-09	3.8E-07
Mo-99	1.7E+03	4.6E-02	2.0E-04	5.4E-15	5.4E-15	2.0E-09	2.7E-06
Tc-99m	2.2E+00	5.9E-05	2.6E-07	7.0E-18	7.0E-18	2.0E-07	3.5E-11
Ru-103	1.0E+02	2.7E-03	1.2E-05	3.2E-16	3.2E-16	9.0E-10	3.5E-07
Rh-103m	3.5E-03	9.5E-08	4.1E-10	1.1E-20	1.1E-20	2.0E-06	5.5E-15
Ru-106	1.4E-01	3.8E-06	1.6E-08	4.4E-19	4.4E-19	2.0E-11	2.2E-08
Rh-106	4.5E-06	1.2E-10	5.3E-13	1.4E-23	1.4E-23	1.0E-09	1.4E-14
Ag-110m	1.0E-01	2.7E-06	1.2E-08	3.2E-19	3.2E-19	1.0E-10	3.2E-09
Sb-124	5.3E+00	1.4E-04	6.2E-07	1.7E-17	1.7E-17	3.0E-10	5.6E-08
Te-129m	1.6E+00	4.3E-05	1.9E-07	5.1E-18	5.1E-18	3.0E-10	1.7E-08
Te-131m	5.5E-01	1.5E-05	6.5E-08	1.7E-18	1.7E-18	1.0E-09	1.7E-09
Te-132	1.4E-01	3.8E-06	1.6E-08	4.4E-19	4.4E-19	9.0E-10	4.9E-10

NAPS COL 12.2-2-A
NAPS ESP COL 11.1-1

Table 12.2-17R Comparison of Airborne Release Concentrations with 10 CFR 20 Limit

Nuclide	Unit 3 Annual Release MBq/yr	Unit 3 Annual Release Ci/yr	Unit 3 Concentration Bq/m^3	Unit 3 Concentration µCi/ml	Units 1, 2 & 3 Concentration µCi/ml	ECL µCi/ml	Units 1, 2, & 3 Fraction of ECL
Cs-134	1.8E+02	4.9E-03	2.1E-05	5.7E-16	5.7E-16	2.0E-10	2.9E-06
Cs-136	1.5E+01	4.1E-04	1.8E-06	4.8E-17	4.8E-17	9.0E-10	5.3E-08
Cs-137	2.7E+02	7.3E-03	3.2E-05	8.6E-16	8.6E-16	2.0E-10	4.3E-06
Cs-138	8.5E-01	2.3E-05	1.0E-07	2.7E-18	2.7E-18	8.0E-08	3.4E-11
Ba-140	7.8E+02	2.1E-02	9.2E-05	2.5E-15	2.5E-15	2.0E-09	1.2E-06
La-140	1.3E+01	3.5E-04	1.5E-06	4.1E-17	4.1E-17	2.0E-09	2.1E-08
Ce-141	2.6E+02	7.0E-03	3.1E-05	8.2E-16	8.2E-16	8.0E-10	1.0E-06
Ce-144	1.3E-01	3.5E-06	1.5E-08	4.1E-19	4.1E-19	2.0E-11	2.1E-08
Pr-144	1.6E-04	4.3E-09	1.9E-11	5.1E-22	5.1E-22	2.0E-07	2.5E-15
W-187	1.3E+00	3.5E-05	1.5E-07	4.1E-18	4.1E-18	1.0E-08	4.1E-10
Np-239	8.3E+01	2.2E-03	9.7E-06	2.6E-16	2.6E-16	3.0E-09	8.8E-08
Total w/o H-3	1.7E+08	4.5E+03	2.0E+01	5.3E-10	1.2E-08	NA	1.8E-01
Total w/ H-3	1.7E+08	4.6E+03	2.0E+01	5.4E-10	1.2E-08	NA	1.8E-01

Note: Concentrations for Units 1 and 2 are based on the activity releases in NAPS UFSAR Table 11.3-2. Effluent concentration limits (ECLs) are from 10 CFR 20, Appendix B, Table 2, Column 1.

NAPS COL 12.2-2-A
NAPS ESP COL 11.1-1

Table 12.2-18aR Airborne Offsite Dose Calculation Bases

NAPS COL 12.2-2-A	Meteorology χ/Q	Table 2.3-16R
NAPS COL 12.2-2-A	Meteorology D/Q	Table 2.3-16R
	Airborne Release Source Term	DCD Table 12.2-16
	Calculation Methodology	RG 1.109
	Computer Code Utilized	GASPAR II (NUREG/CR-4653)
	Individual Consumption Rates	Table E-5 of RG 1.109
	Misc. Calculation Inputs (other than RG 1.109 default values):	
NAPS COL 12.2-2-A	Midpoint of plant operating life	20 years
NAPS COL 12.2-2-A	Fraction of year that leafy vegetables are grown	0.5
NAPS COL 12.2-2-A	Fraction of year that animals graze on pasture	0.67
NAPS COL 12.2-2-A	Fraction of daily feed that is pasture grass when the animal grazes on pasture	1.0
NAPS COL 12.2-2-A	Animal milk considered for milk pathway	None – no milk animal within 8 km (5 mi)
NAPS COL 12.2-2-A	Annual Average Doses from Airborne Releases	Table 12.2-18bR

NAPS COL 12.2-2-A
NAPS ESP COL 11.1-1
NAPS ESP VAR 12.2-1

Table 12.2-18bR Gaseous Pathway Doses to the MEI (mrem/yr)

		ESP			Unit 3		
Location	**Pathway**	**Total Body**	**Thyroid**	**Skin**	**Total Body**	**Thyroid**	**Skin**
Site Boundary (1416 m (0.88 mi) ESE for ESP-ER and FSAR)	Plume	2.1E+00	NA	6.2E+00	1.6E+00	**1.6E+00**	4.0E+00
	Inhalation						
	Adult	3.0E-01	1.6E+00	NA	9.1E-03	6.8E-01	NA
	Teen	3.1E-01	2.0E+00	NA	9.7E-03	8.9E-01	NA
	Child	2.7E-01	2.3E+00	NA	9.1E-03	1.1E+00	NA
	Infant	1.6E-01	2.0E+00	NA	5.5E-03	9.8E-01	NA
Nearest Garden (1513 m (0.94 mi) NE for ESP-ER; 1191 m (0.74 mi) ESE for FSAR)	Vegetable						
	Adult	4.4E-01	4.9E+00	NA	3.7E-01	4.0E+00	NA
	Teen	5.7E-01	6.6E+00	NA	**5.8E-01**	5.5E+00	NA
	Child	1.1E+00	1.3E+01	NA	**1.3E+00**	1.1E+01	NA
Nearest Residence (1545 m (0.96 mi) NNE for ESP-ER; 1191 m (0.74 mi) ESE for FSAR)	Plume	1.4E+00	NA	4.0E+00	3.2E-01	**3.2E-01**	6.5E-01
	Inhalation						
	Adult	2.0E-01	1.0E+00	NA	9.9E-03	7.2E-01	NA
	Teen	2.0E-01	1.3E+00	NA	1.0E-02	9.3E-01	NA
	Child	1.8E-01	1.5E+00	NA	9.6E-03	1.1E+00	NA
	Infant	1.0E-01	1.3E+00	NA	5.8E-03	1.0E+00	NA
Nearest Meat Cow (2205 m (1.37 mi) SE for ESP-ER; 1191 m (0.74 mi) ESE for FSAR)	Meat						
	Adult	6.7E-02	1.5E-01	NA	**1.3E-01**	**2.6E-01**	NA
	Teen	4.9E-02	1.1E-01	NA	**1.1E-01**	**2.0E-01**	NA
	Child	7.9E-02	1.7E-01	NA	**2.0E-01**	**3.4E-01**	NA

NAPS COL 12.2-2-A
NAPS ESP COL 11.1-1
NAPS ESP VAR 12.2-1

Table 12.2-18bR Gaseous Pathway Doses to the MEI (mrem/yr)

		ESP			Unit 3		
Location	**Pathway**	**Total Body**	**Thyroid**	**Skin**	**Total Body**	**Thyroid**	**Skin**
Nearest Garden/ Residence/ Meat Cow (Varies for ESP-ER; 1191 m (0.74 mi) ESE for FSAR)	All						
	Adult	1.6E+00	4.9E+00	4.0E+00	8.3E-01	**5.3E+00**	6.5E-01
	Teen	1.6E+00	6.6E+00	4.0E+00	1.0E+00	**7.0E+00**	6.5E-01
	Child	1.6E+00	1.3E+01	4.0E+00	**1.9E+00**	1.3E+01	6.5E-01
	Infant	1.5E+00	1.3E+00	4.0E+00	3.3E-01	**1.4E+00**	6.5E-01

Notes:

1. There are no infant doses for the vegetable and meat pathways because infants do not consume these foods.
2. “NA” denotes “not applicable.”
3. 1 mrem = 0.01 msv
4. For Unit 3, the doses shown for “nearest garden/residence/meat cow” location are the sum of garden, residence, and meat cow doses at 1191m ESE. For ESP, these doses are the maximum of garden, residence, and meat cow doses at 1513m NE, 1545 m NNE, and 2205m SE, respectively. The site boundary and residence plume doses include ground shine contribution.
5. The maximum (child) bone dose for Unit 3 from all gaseous effluent pathways is shown in Table 12.2-203.

Table 12.2-19bR Comparison of Annual Liquid Release Concentrations with 10 CFR 20 Limit

NAPS COL 12.2-3-A
NAPS ESP COL 11.1-1
NAPS ESP VAR 12.2-3

	Unit 3 Annual Release		Unit 3 Concentration		Units 1, 2 & 3 Concentration	ECL	Units 1, 2, & 3 Fraction of ECL
Nuclide	MBq/yr	Ci/yr	Bq/ml	µCi/ml	µCi/ml	µCi/ml	
I-131	1.55E+02	4.2E-03	7.8E-07	2.1E-11	5.6E-08	1.0E-06	5.6E-02
I-132	3.03E+01	8.2E-04	1.5E-07	4.1E-12	8.5E-09	1.0E-04	8.5E-05
I-133	7.77E+02	2.1E-02	4.1E-06	1.1E-10	6.2E-08	7.0E-06	8.9E-03
I-134	1.48E+00	4.0E-05	7.4E-09	2.0E-13	1.2E-09	4.0E-04	3.0E-06
I-135	2.00E+02	5.4E-03	1.0E-06	2.7E-11	3.6E-09	3.0E-05	1.2E-04
H-3	5.18E+05	1.4E+01	4.4E-03	1.2E-07	5.6E-06	1.0E-03	5.6E-03
Na-24	**1.89E+02**	**5.1E-03**	9.6E-07	2.6E-11	2.6E-11	5.0E-05	5.1E-07
P-32	1.55E+01	4.2E-04	7.8E-08	2.1E-12	2.1E-12	9.0E-06	2.3E-07
Cr-51	4.81E+02	1.3E-02	2.4E-06	6.6E-11	8.9E-11	5.0E-04	1.8E-07
Mn-54	5.92E+00	1.6E-04	3.7E-08	1.0E-12	4.0E-11	3.0E-05	1.3E-06
Mn-56	4.81E+01	1.3E-03	2.4E-07	6.5E-12	6.5E-12	7.0E-05	9.3E-08
Fe-55	8.51E+01	2.3E-03	6.3E-07	1.7E-11	1.7E-11	1.0E-04	1.7E-07
Fe-59	2.59E+00	7.0E-05	1.3E-08	3.6E-13	2.6E-11	1.0E-05	2.6E-06
Co-58	1.63E+01	4.4E-04	8.9E-08	2.4E-12	7.4E-10	2.0E-05	3.7E-05
Co-60	3.33E+01	9.0E-04	2.7E-07	7.2E-12	6.7E-11	3.0E-06	2.2E-05
Cu-64	**4.81E+02**	**1.3E-02**	2.4E-06	6.5E-11	6.5E-11	2.0E-04	3.3E-07
Zn-65	1.67E+01	4.5E-04	1.0E-07	2.8E-12	2.8E-12	5.0E-06	5.6E-07
Zn-69m	**3.40E+01**	**9.2E-04**	1.7E-07	4.6E-12	4.6E-12	6.0E-05	7.7E-08
Br-83	**3.33E+00**	**9.0E-05**	1.7E-08	4.5E-13	4.5E-13	9.0E-04	5.0E-10

Table 12.2-19bR Comparison of Annual Liquid Release Concentrations with 10 CFR 20 Limit

NAPS COL 12.2-3-A
NAPS ESP COL 11.1-1
NAPS ESP VAR 12.2-3

	Unit 3 Annual Release		Unit 3 Concentration		Units 1, 2 & 3 Concentration	ECL	Units 1, 2, & 3 Fraction of ECL
Nuclide	**MBq/yr**	**Ci/yr**	**Bq/ml**	**µCi/ml**	**µCi/ml**	**µCi/ml**	
Sr-89	8.14E+00	2.2E-04	4.4E-08	1.2E-12	1.1E-10	8.0E-06	1.4E-05
Sr-90	7.40E-01	2.0E-05	6.3E-09	1.7E-13	1.2E-11	5.0E-07	2.4E-05
Sr-91	**4.44E+01**	**1.2E-03**	2.2E-07	6.0E-12	2.5E-11	2.0E-05	1.3E-06
Y-91	5.18E+00	1.4E-04	2.7E-08	7.4E-13	1.3E-10	8.0E-06	1.6E-05
Sr-92	1.07E+01	2.9E-04	5.6E-08	1.5E-12	1.5E-12	4.0E-05	3.6E-08
Y-92	**4.07E+01**	**1.1E-03**	2.0E-07	5.5E-12	5.5E-12	4.0E-05	1.4E-07
Y-93	**4.44E+01**	**1.2E-03**	2.2E-07	6.0E-12	6.0E-12	2.0E-05	3.0E-07
Zr-95	7.40E-01	2.0E-05	4.1E-09	1.1E-13	2.1E-11	2.0E-05	1.1E-06
Nb-95	7.40E-01	2.0E-05	3.7E-09	1.0E-13	2.2E-11	3.0E-05	7.4E-07
Mo-99	1.11E+02	3.0E-03	5.6E-07	1.5E-11	9.9E-08	2.0E-05	5.0E-03
Tc-99m	**2.04E+02**	**5.5E-03**	1.0E-06	2.8E-11	8.5E-08	1.0E-03	8.5E-05
Ru-103	1.48E+00	4.0E-05	7.8E-09	2.1E-13	2.1E-13	3.0E-05	6.9E-09
Ru-105	**6.29E+00**	**1.7E-04**	3.1E-08	8.5E-13	8.5E-13	7.0E-05	1.2E-08
Te-129m	3.33E+00	9.0E-05	1.7E-08	4.6E-13	4.6E-13	7.0E-06	6.6E-08
Te-131m	3.70E+00	1.0E-04	1.9E-08	5.0E-13	5.0E-13	8.0E-06	6.3E-08
Te-132	7.40E-01	2.0E-05	3.7E-09	1.0E-13	4.8E-09	9.0E-06	5.3E-04
Cs-134	2.52E+01	6.8E-04	1.9E-07	5.0E-12	1.8E-08	9.0E-07	2.0E-02
Cs-136	1.52E+01	4.1E-04	7.8E-08	2.1E-12	2.6E-09	6.0E-06	4.3E-04
Cs-137	6.66E+01	1.8E-03	5.6E-07	1.5E-11	1.2E-07	1.0E-06	1.2E-01

Table 12.2-19bR Comparison of Annual Liquid Release Concentrations with 10 CFR 20 Limit

NAPS COL 12.2-3-A
NAPS ESP COL 11.1-1
NAPS ESP VAR 12.2-3

	Unit 3 Annual Release		Unit 3 Concentration		Units 1, 2 & 3 Concentration	ECL	Units 1, 2, & 3 Fraction of ECL
Nuclide	**MBq/yr**	**Ci/yr**	**Bq/ml**	**µCi/ml**	**µCi/ml**	**µCi/ml**	
Ba-139	**1.48E+00**	**4.0E-05**	7.4E-09	2.0E-13	2.0E-13	2.0E-04	1.0E-09
Ba-140	3.03E+01	8.2E-04	1.5E-07	4.1E-12	9.6E-11	8.0E-06	1.2E-05
Ce-141	2.59E+00	7.0E-05	1.3E-08	3.6E-13	3.6E-13	3.0E-05	1.2E-08
La-142	**1.11E+00**	**3.0E-05**	5.6E-09	1.5E-13	1.5E-13	1.0E-04	1.5E-09
Ce-143	1.11E+00	3.0E-05	5.6E-09	1.5E-13	1.5E-13	2.0E-05	7.5E-09
Pr-143	3.33E+00	9.0E-05	1.7E-08	4.5E-13	4.5E-13	2.0E-05	2.3E-08
W-187	**8.88E+00**	**2.4E-04**	4.4E-08	1.2E-12	1.2E-12	3.0E-05	4.0E-08
Np-239	4.07E+02	1.1E-02	2.0E-06	5.5E-11	5.5E-11	2.0E-05	2.8E-06
Total w/o H-3	3.62E+03	9.80E-02	1.9E-05	5.1E-10	4.6E-07	NA	2.1E-01
Total w/ H-3	5.22E+05	1.41E+01	4.5E-03	1.2E-07	6.1E-06	NA	2.2E-01

Note: Concentrations for Units 1 and 2 are obtained from NAPS UFSAR Table 11.2-14. ECLs are from 10 CFR 20, Appendix B, Table 2, Column 2.

NAPS COL 12.2-3-A
NAPS ESP COL 11.1-1

Table 12.2-20aR Liquid Pathway Offsite Dose Calculation Bases

	Calculation Methodology	RG 1.109
	Computer Code Utilized	LADTAP II (NUREG/CR-4013)
	Individual Consumption/Exposure Rates	Table E-5 of RG 1.109
	Site Water Type	Freshwater
NAPS COL 12.2-3-A	Liquid Effluent Discharge Rate	3.8E+02 liters/min (0.223 ft^3/sec)
NAPS COL 12.2-3-A	Shore-Width Factor	0.3
NAPS COL 12.2-3-A	Dilution Factor	1000
NAPS COL 12.2-3-A	Transit times from discharge to the receiving water body to exposure location	All pathways: instantaneous
NAPS COL 12.2-3-A	Irrigation rate	None - lake water is not used for irrigation
NAPS COL 12.2-3-A	Fraction of year that leafy vegetables are grown	Not used in liquid pathway dose calculation
NAPS COL 12.2-3-A	Fraction of year that animals graze on pasture	Not used in liquid pathway dose calculation
NAPS COL 12.2-3-A	Fraction of daily feed that is pasture grass when the animal grazes on pasture	Not used in liquid pathway dose calculation
NAPS COL 12.2-3-A	Animal milk considered for milk pathway	Not used in liquid pathway dose calculation
NAPS COL 12.2-3-A	Liquid Pathway Offsite Annual Doses	Table 12.2-20bR

NAPS COL 12.2-3-A
NAPS ESP COL 11.1-1

Table 12.2-20bR Liquid Pathway Doses from Unit 3 for Maximally Exposed Individuals at Lake Anna

	ESP-ER Dose (mrem/yr)			Unit 3 Dose (mrem/yr)		
Pathway	Total Body	Thyroid	Bone	Total Body	Thyroid	Bone
Fish	5.1E-01	NA	2.3E-00	7.8E-02	NA	1.2E-00
Invertebrate	6.6E-02	NA	1.5E-01	8.3E-03	NA	6.5E-02
Drinking	2.0E-01	6.5E-01	2.7E-02	4.1E-03	1.8E-01	5.6E-03
Shoreline	3.0E-02	3.0E-02	3.0E-02	3.0E-03	3.0E-03	3.0E-03
Swimming	3.2E-04	3.2E-04	3.2E-04	1.2E-04	1.2E-04	1.2E-04
Boating	4.0E-04	4.0E-04	4.0E-04	1.5E-04	1.5E-04	1.5E-04
Total	8.1E-01	6.8E-01	2.5E-00	9.4E-02	1.8E-01	1.3E-00
Age group receiving maximum dose	Adult	Infant	Child	Adult	Infant	Child

Notes: 1. Bone of the child is the organ receiving the maximum dose.
2. There are no infant doses for the fish and invertebrate pathways because infants do not consume these foods.
3. “NA” denotes “not applicable.”
4. 1 mrem = 0.01 mSv

NAPS COL 12.2-2-A
NAPS
ESP COL 11.1-1

Table 12.2-201 Comparison of Annual Doses to the MEI from Gaseous Effluents Per Unit

Type of Dose	Location	ESP (Single Unit)	Unit 3	10 CFR 50 Limit
Gamma Air (mrad/yr)	Site Boundary	3.2	2.2	10
Beta Air (mrad/yr)	Site Boundary	4.8	2.5	20
Total Body (mrem/yr)	Site Boundary	2.4	1.6	5
Skin (mrem/yr)	Site Boundary	6.2	4.0	15
Iodines and Particulates - Thyroid (mrem/yr)	Garden/ Residence/ Meat Cow	12	11	15

1 mrad = 0.01 mGy
1 mrem = 0.01 mSv

NAPS COL 12.2-3-A
NAPS ESP COL 11.1-1

Table 12.2-202 Comparison of Annual Doses to MEI from Liquid Effluents Per Unit

Type of Dose	Location	ESP (Single Unit)	Unit 3	10 CFR 50 Limit
Total Body (mrem/yr)	Lake Anna	0.81	0.094	3
Bone (mrem/yr)	Lake Anna	2.5	1.3	10

1 mrem = 0.01 msv

NAPS COL 12.2-2-A
NAPS COL 12.2-3-A
NAPS ESP COL 11.1-1
NAPS ESP VAR 12.2-4

Table 12.2-203 Comparison of Site Doses to the MEI

Type of Dose	ESP Site Total(1)	Unit 3 (ESBWR)			Existing Units(2)	Site Total(3)	40 CFR 190 Limit
		Liquid	Gaseous	Total (5)			
Total Body (mrem/yr)	6.8	0.094	1.9	2.0	**5.0**	**6.9**	25
Thyroid (mrem/yr)	27	0.18	13	13	**5.1**	18	75
Bone (mrem/yr)	12	1.3	8.0	9.2	**5.1**	**14**	25

Notes:

1. The ESP site total doses are for two new units and two existing units, and do not include a dose contribution from the ISFSI.
2. The doses from existing units include ISFSI contribution and an assumed dose of 1 mrem/yr due to direct radiation from the existing units.
3. This site total dose includes the Unit 3 total dose and the dose from the existing units.
4. 1 mrem = 0.01 mSv
5. Unit 3 total annual doses include a Turbine Building skyshine contribution of less than 1.66E-04 mrem/yr.

NAPS COL 12.2-2-A
NAPS COL 12.2-3-A
NAPS ESP COL 11.1-1

Table 12.2-204 Collective Total Body (Population) Doses Within 50 Miles

Units in person-rem/yr

	ESP (Single Unit)	Unit 3
Liquid	8.6	1.0
Noble Gases (Gaseous)	3.5	1.5
Iodines and Particulates (Gaseous)	1.4	0.88
H-3 and C-14 (Gaseous)	14	5.3
Gaseous Total	19	7.7
Total	28	8.7

Notes:
1. 1 rem = 0.01 Sv
2. ESP doses are based on data from ESP-ER Tables 2.5-8, 5.4-1, and 5.4-3.
3. The corresponding collective thyroid doses for Unit 3 are 0.69 person-rem/year from liquid effluents and 28 person-rem/year from gaseous effluents.
4. The long-term χ/Q and D/Q values used in deriving Unit 3 collective doses from routine gaseous effluent releases within 50 miles of the plant are shown in Tables 2.3-208 to 2.3-215.

12.3 Radiation Protection

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

12.3.1.3 **Radiation Zoning**

Replace the last sentence with the following.

STD COL 12.3-3-H Access to "Very High Radiation Areas" is discussed in Section 12.5.

12.3.4 **Area Radiation and Airborne Radioactivity Monitoring Instrumentation**

Replace the last bullet with the following.

STD COL 12.3-2-A The radiation instrumentation that monitors airborne radioactivity is classified as nonsafety-related. Airborne radiation monitoring operational considerations, such as the procedures for operation and calibration of the monitors, as well as the placement of the portable monitors, are discussed in Section 12.5.

12.3.7 **COL Information**

12.3-2-A **Operational Considerations**

STD COL 12.3-2-A This COL item is addressed in Section 12.3.4.

12.3-3-H **Controlled Access**

STD COL 12.3-3-H This COL item is addressed in Section 12.3.1.3.

12.4 Dose Assessment

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

NAPS SUP 12.4-1

12.4.7.1 **Annual Doses to Construction Workers**

Doses to construction workers were addressed in ESP-ER Section 4.5 and associated impacts were resolved as SMALL in FEIS Section 4.9.

The ESP-ER analysis has been reviewed to evaluate the following more recent information:

- The current locations and readings for TLDs located closest to the Unit 3 site.

- The most recent effluent release data for Units 1 and 2 as reported in the 2006 Annual Radioactive Effluent Release Report (Reference 12.4-201).
- Spent fuel cask types planned for the onsite Independent Spent Fuel Storage Installation have changed.
- The estimated peak number of construction workers is now 2500-3000 (versus 5000 in the ESP-ER).

Based on the results of this review, it is concluded that the 120 person-rem calculated in the ESP-ER remains a conservative estimate of the maximum annual collective dose to the construction work force.

12.4.9 **References**

12.4-201 Annual Radioactive Effluent Release Report, January 1, 2006 – December 31, 2006, Dominion Virginia Power.

12.5 Operational Radiation Protection Program

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

12.5.3 **Operational Considerations**

Replace this section with the following.

STD COL 12.5-1-A
STD COL 12.5-2-A
STD COL 12.5-3-A

The operational program for radiation protection is addressed in Appendix 12BB.

12.5.4 **COL Information**

12.5-1-A **Equipment, Instrumentation, and Facilities**

STD COL 12.5-1-A This COL item is addressed in Appendix 12BB.

12.5-2-A **Compliance with 10 CFR Part 50.34(f)(2)(xxvii) and NUREG-0737 Item III.D.3.3**

STD COL 12.5-2-A This COL item is addressed in Appendix 12BB.

12.5-3-A **Radiation Protection Program**

STD COL 12.5-3-A This COL item is addressed in Appendix 12BB.

12.6 Minimization of Contamination and Radwaste Generation

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

12.6.1 Minimization of Contamination to Facilitate Decommissioning

Add the following at the end of this section.

STD SUP 12.6-1

In addition to design features, measures are implemented in operating procedures to minimize contamination. Appendix 12BB establishes contamination control measures to ensure compliance with 10 CFR 20.1406. Practical measures to prevent the spread of contamination are employed, including:

- Engineering controls, such as portable ventilation or filtration units to reduce concentrations of radioactivity in air or fluids, are used where practical
- Criteria for selecting tools, material, and equipment for use in contaminated areas include minimizing the use of porous or other materials that are difficult to decontaminate
- Leaks and spills are contained promptly and repaired or cleaned up as soon as practical
- Containments, caches, and enclosures are used during maintenance, repairs, and testing, when practical, to contain spills or releases
- Contaminated tools and equipment are segregated from clean tools and equipment
- Potentially contaminated systems, equipment, and components are surveyed for the presence of contamination when opened or prior to removal
- Procedures ensure that equipment performs and is operated in accordance with the design requirements
- Temporary and permanent design modifications require compensatory measures be taken to prevent and limit the spread of contamination

Appendix 12A Calculation of Airborne Radionuclides

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

Appendix 12B Calculation of Airborne Releases

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

STD SUP 12.1-1

Appendix 12AA ALARA Program

NEI 07-08, Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA), which is currently under review by the NRC staff, is incorporated by reference. (Reference 12AA-201)

12AA.1 References

12AA-201 Nuclear Energy Institute (NEI), Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA), NEI 07-08.

STD COL 12.1-1-A
STD COL 12.1-2-A
STD COL 12.1-3-A
STD COL 12.1-4-A
STD COL 12.5-1-A
STD COL 12.5-2-A
STD COL 12.5-3-A

Appendix 12BB Radiation Protection

NEI 07-03, Generic FSAR Template Guidance for Radiation Protection Program Description, which is currently under review by the NRC staff, is incorporated by reference with the following supplemental information. (Reference 12BB-201)

12.5.2.4 Radiation Protection Technicians

Delete the third paragraph.

12.5.3.1 Facilities

Delete the first and second paragraphs.

12.5.3.2 Monitoring Instrumentation and Equipment

Delete the third paragraph.

12.5.3.3 Personal Protective Clothing and Equipment

Delete the last sentence in the first paragraph.

12.5.4.2 Methods to Maintain Exposures ALARA

Delete the second paragraph.

12.5.4.4 Access Control

Isometric drawings of the Very High Radiation Areas (VHRA) are included in DCD Section 12.3.

Physical access controls include postings, barricades, physical barriers, and the use of locks that are keyed so only keys designated as VHRA can open the locks. Additionally, entry into a VHRA is allowed only with a specific (Special) radiation work permit.

12.5.4.12 Quality Assurance

Replace the bracketed text in the first paragraph with Section 17.5.

12BB.1 References

12BB-201 Nuclear Energy Institute (NEI), Generic FSAR Template Guidance for Radiation Protection Program Description, NEI 07-03.

Chapter 13 Conduct of Operations

The introductory paragraph of this chapter of the referenced DCD is incorporated by reference with no departures or supplements.

13.1 Organizational Structure of Applicant

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

DCD Section 13.1.1, Combined License Information, is renumbered in this FSAR as Section 13.1.4 for administrative purposes to allow section numbering to be consistent with RG 1.206 and the Standard Review Plan.

Replace the first paragraph with the following.

NAPS COL 13.1-1-A

This section describes the organization of Unit 3. The organizational structure is described in this section and is consistent with the Human System Interface (HSI) design assumptions used in the design of the ESBWR as described in DCD Chapter 18. The organizational structure is consistent with the ESBWR HFE design requirements and complies with the requirements of 10 CFR 50.54(i) through (m).

13.1.1 Management and Technical Support Organization

Dominion has over 35 years of experience in the design, construction, and operation of nuclear generating stations. Dominion and its affiliates currently operates seven nuclear units at four sites located in Virginia, Connecticut, and Wisconsin.

Corporate offices provide support for the nuclear stations. Figure 13.1-205 illustrates the relationship of the nuclear organization to other divisions of Dominion. This support includes executive level management to provide strategic and financial support for plant initiatives, coordination of functional efforts division-wide, and functional level management in areas such as training, security, emergency planning, and engineering analysis.

Figure 13.1-204 provides a high-level illustration of the nuclear organization. More detailed charts and position descriptions, including qualification requirements and staffing numbers for corporate support staff, are maintained in corporate offices.

Changes to the organization described herein are reviewed under the provisions of 10 CFR 50.54(a) to ensure that any reduction in commitments in the QAPD (as accepted by the NRC) are submitted to, and approved by the NRC, prior to implementation.

13.1.1.1 Design, Construction, and Operating Responsibilities

The chief nuclear officer (CNO) has overall responsibility for functions involving planning, design, construction, and operation of Dominion's nuclear units. Line responsibilities for those functions are passed to the executives in charge of nuclear operations, engineering and technical services, planning, development, and oversight, who maintain direct control of nuclear plant activities.

The first priority and responsibility of each member of the nuclear staff throughout the life of the plant is nuclear safety. Decision making for station activities is performed in a conservative manner with expectations of this core value regularly communicated to appropriate personnel by management interface, training, and station directives.

Lines of authority and communication clearly and unambiguously establish that utility management directs the project.

At key project milestones, including beginning of construction, fuel load, and commercial operation, senior management will determine if there are sufficient numbers of qualified personnel available to move the project forward.

The construction management organization is shown in Figure 13.1-201.

13.1.1.1.1 Design and Construction Responsibilities

This section is included in Appendix 13AA for future designation as historical information.

13.1.1.2 Technical Support for Plant Operations

This section describes the functional groups that will be activated before fuel load. The site vice president will establish the organization of managers, functional managers, supervisors, and staff sufficient to perform required functions for support of safe plant operation. These functions include the following:

- Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgical and material, and instrumentation and controls engineering

- Plant chemistry
- Radiation protection
- Fueling and refueling operations support
- Maintenance support
- Operations support
- Quality assurance
- Training
- Safety review
- Fire protection
- Emergency organization
- Outside contractual assistance

In the event that station personnel are not qualified to deal with a specific problem, the services of qualified individuals from other functions within the company or outside consultants are engaged. Figure 13.1-204 illustrates the nuclear operating organization. Table 13.1-201 shows the estimated number of positions required for each function.

13.1.1.2.1 **Engineering**

The site engineering department consists of system engineering, design engineering, component engineering, project engineering, and engineering programs. These groups are responsible for performing the classical design activities as well as providing engineering expertise for programs such as reactor engineering, fire protection, inservice inspection (ISI), inservice testing (IST), snubbers, and maintenance rule. Corporate engineering provides support for engineering projects, safety and engineering analysis, and nuclear fuels engineering. They are responsible for probabilistic safety assessment and other safety issues, plant system reliability analysis, performance and technical support, core management, and periodic reactor testing.

Each of the site engineering groups has a functional manager who reports to the site director of nuclear engineering.

The engineering organization is responsible for:

- Support of plant operations in the engineering areas of mechanical, structural, electrical, thermal-hydraulic, metallurgy and materials, electronic, instrument and control, and fire protection. Priorities for support activities are established based on input from the plant manager with emphasis on issues affecting safe operation of the plant.

- Support of procurement, chemical and environmental analysis, and maintenance activities in the plant as requested by the plant manager
- Performance of design engineering of plant modifications
- Maintaining the design basis by updating the record copy of design documents as necessary to reflect the actual as-built configuration of the plant
- Accident and transient analyses
- Human Factors Engineering design process

Reactor engineering, led by the functional manager in charge of reactor engineering, provides technical assistance in the areas of core operations, core thermal limits, and core thermal hydraulics.

Design work may be contracted to and performed by outside companies in accordance with Appendix 17AA, Sections 2 and 2.2.

13.1.1.2.2 **Plant Chemistry**

A chemistry program is established to monitor and control the chemistry of various plant systems such that corrosion of components and piping is minimized and radiation from corrosion by-products is kept to levels that allow operations and maintenance with radiation doses as low as is reasonably achievable.

The functional manager in charge of chemistry is responsible for maintaining chemistry programs and for monitoring and maintaining the water chemistry of plant systems. The staff of the chemistry department consists of laboratory technicians, support personnel, and supervisors who report to the functional manager in charge of chemistry.

13.1.1.2.3 **Radiation Protection**

A radiation protection (RP) program is established to protect the health and welfare of the surrounding public and personnel working at the plant. The RP program is described in Chapter 12.

The RP department is staffed by radiation protection technicians, support personnel, and supervisors who report to the radiation protection manager. To provide sufficient organizational freedom from operating pressures, the radiation protection manager reports directly to the director responsible for facility safety and licensing.

Personnel resources of the RP organization are shared between units. A single management organization oversees RP for the units.

13.1.1.2.4 **Fueling and Refueling Operations Support**

The function of fueling and refueling is performed by a combination of personnel from various departments including operations, maintenance, radiation protection, engineering, and reactor technology vendor or other contractor staff. Initial fueling is a function of the startup management organization discussed in Appendix 13AA. Refueling operations are a function of the operations organization.

13.1.1.2.5 **Maintenance Support**

The maintenance department includes mechanical maintenance, electrical maintenance, and instrumentation and control (I&C) groups. Each group includes supervisors, foremen, and technicians in sufficient numbers to provide for the safe and efficient operation of the plant during all phases of plant life.

In support of maintenance activities, planners, schedulers, and parts specialists prepare work packages, acquire proper parts, and develop procedures that provide for the successful completion of maintenance tasks. Maintenance tasks are integrated into the station schedule for evaluation of operating or safe shutdown risk elements and to provide for efficient and safe performance. Functional managers in charge of planning and scheduling report to outage and planning management.

13.1.1.2.6 **Operations Support**

The operations support function is provided under the direction of the operations manager, and includes the following programs:

- Operations procedures
- Operations surveillances
- Equipment tagging preparation
- Fuel handling

13.1.1.2.7 **Quality Assurance**

Safety-related activities associated with the operation of the plant are governed by the quality assurance (QA) program established in Chapter 17. QA includes:

- Maintenance of the QAPD
- Coordinating the development of audit schedules
- Audit, surveillance, and evaluation of Nuclear Division suppliers
- Quality control (QC) inspection/testing activities

QA management is independent of the station management line organization. The manager of QA reports to the corporate-stationed director of oversight.

Personnel resources of the QA organization are shared between units. A single management organization oversees the QA group for the site.

13.1.1.2.8 **Training**

The training department is responsible for providing training programs that are established, maintained, and implemented in accordance with applicable plant administrative directives, regulatory requirements, and company operating policies so that station personnel can meet the performance requirements of their jobs in operations, maintenance, technical support, emergency response, and other areas. The training department's responsibilities encompass operator initial license training, requalification training, and plant staff training as well as the plant access training (general employee training) course and radiation worker training. To maintain independence from operating pressures, the manager of training reports to the corporate training organization. Nuclear plant training programs are described in Section 13.2.

To the extent practicable given the differences between plant designs, personnel resources of the training department are shared between units. A single management organization provides oversight of station training activities.

13.1.1.2.9 **Safety Review**

Review and audit activities are addressed in Chapter 17.

Oversight of station programs, procedures, and activities is performed by the Facilities Safety Review Committee (FSRC), a corporate safety review committee, and Station Nuclear Safety (SNS) department, which is responsible for corrective actions and assessments. The supervisor in charge of SNS ultimately reports to the site vice president.

Personnel resources of the SNS organization are shared between units. A single management organization oversees the site SNS organization.

In the event of an unplanned reactor trip or significant power reduction, the FSRC is responsible for determining the circumstances, analyzing the cause, and determining that operations can proceed safely before the reactor is returned to power.

13.1.1.2.10 **Fire Protection**

The station is committed to maintaining a fire protection program as described in DCD Section 9.5.1.15. The site executive in charge of plant management has overall responsibility for the fire protection program. Assigning the responsibility at that level provides the authority to obtain the resources and assistance necessary to meet fire protection program objectives, resolve conflicts, and delegate appropriate responsibility to fire protection staff. Fire protection for the facility is organized and administered by fire protection engineer. The fire protection engineer is responsible for development and implementation of the fire protection program, including development of fire protection procedures, and inspections of fire protection systems and functions. The fire protection engineer reports to the functional manager in charge of engineering programs. Functional descriptions for all responsible positions are included in appropriate procedures. Station personnel are responsible for adhering to the fire protection/prevention requirements detailed in Section 9.5.1. The fire brigade is described in Section 13.1.2.1.5.

During construction:

- The site construction executive is ultimately responsible for fire protection on Unit 3.
- Construction workers will receive fire protection training as part of their indoctrination to the site.
- Periodic fire drills will be conducted on Unit 3.

13.1.1.2.11 **Emergency Organization**

The emergency organization is a matrixed organization composed of personnel who have the experience, training, knowledge, and ability necessary to implement actions to protect the public in the case of emergencies. Managers and station personnel assigned to positions in the emergency organization are responsible for supporting the emergency preparedness organization and the emergency plan as required. The staff members of the emergency planning organization administer and orchestrate drills and training to maintain qualification of station staff members, and develop procedures to guide and direct the emergency organization during an emergency. The emergency preparedness manager reports to the corporate-stationed executive in charge of emergency planning via the corporate emergency manager. The site emergency plan organization is described in the Emergency Plan.

13.1.1.2.12 **Outside Contractual Assistance**

Contract assistance with vendors and outside suppliers is provided by the materials, procurement, and contracts organization. The functional manager in charge of materials, procurement, and contracts reports to the corporate stationed senior manager in charge of materials, purchasing, and contracts.

Resources and management of the materials, procurement, and contracts organization are shared between units.

13.1.1.3 **Organizational Arrangement**

Organizational arrangement for corporate offices and site organizations reporting directly to corporate offices is presented below.

13.1.1.3.1 **Executive/Management Organization**

Executive management is ultimately responsible for execution of activities and functions for Unit 3. Executive management establishes expectations such that a high level of quality, safety, and efficiency is achieved in aspects of plant operations and support activities through an effective management control system and an organization selected and trained to meet the above expectations. The executives with direct line of authority for activities associated with the design, construction, and operation of the plant are shown in Figure 13.1-204. Responsibilities of those executives are discussed below.

13.1.1.3.1.1 **Chief Nuclear Officer**

The CNO has the ultimate responsibility for the safe and reliable operation of each nuclear station owned and/or operated by the utility. It is the responsibility of the CNO to provide guidance and direction such that safety-related activities under his/her direction including engineering, construction, operations, operations support, maintenance, and planning are performed following the guidelines of the quality assurance program.

The CNO delegates authority and responsibility for operation and support of the site through the senior vice president of nuclear operations, the vice president of engineering, the vice president of support services, the vice president of nuclear development, and the director of nuclear oversight. The CNO has no ancillary responsibilities that might detract attention from nuclear safety matters.

13.1.1.3.1.2 **Senior Vice President of Nuclear Operations**

The senior vice president of nuclear operations is responsible for the operation of all nuclear plants owned and/or managed by the utility. The senior vice president of nuclear operations maintains direct control of nuclear plant operations through the site vice president. The senior vice president of nuclear operations reports to the CNO.

13.1.1.3.1.3 **Vice President of Nuclear Engineering**

The vice president of nuclear engineering is responsible for engineering activities associated with operating nuclear plants in the system. The vice president of nuclear engineering performs these functions through managers who are responsible for the functions and programs discussed in Section 13.1.1.2.1. The vice president of nuclear engineering reports to the CNO.

13.1.1.3.1.4 **Vice President of Support Services**

The vice president of support services is responsible for ensuring that nuclear regulatory requirements for operating plants are implemented, and for maintaining lines of communication with the nuclear regulatory authority. The vice president of support services is also responsible for the operating plant support functions of emergency planning, training and development, and security. The direct reports of the vice president of support services include managers responsible for security, training, emergency preparedness, and licensing for the operating plants. The vice president of support services reports to the CNO.

13.1.1.3.1.5 **Vice President of Nuclear Development**

The vice president of nuclear development is responsible for the preparation, submission, and defense of license applications for new nuclear units before the nuclear regulatory authority and for the implementation of regulatory requirements and license conditions upon issuance of the license up to commencement of commercial operations. The vice president of nuclear development is also responsible for engineering oversight and project activities, and for site activities associated with new nuclear units prior to commencement of commercial operations. The direct reports to the vice president of nuclear development include managers responsible for new nuclear projects, new nuclear plant, organizational effectiveness, and licensing and regulatory interface.

13.1.1.3.1.6 **Director of Nuclear Oversight**

The director of nuclear oversight is responsible for the verification of effective company and supplier QA program development, documentation, and implementation. This position is independent of cost and scheduling concerns associated with construction, operations, maintenance, modification, and decommissioning activities for performing quality assurance program verification. Where implementation of any or all of these functions is delegated to suppliers, procedures require the establishment of interface documents including defining lines of communication and authorities as appropriate for the delegated functions. However, this senior management position retains responsibility for the scope and effective implementation of the quality assurance program for those functions.

This management position has the necessary authority and responsibility for verifying quality achievement; identifying quality problems, recommending solutions and verifying implementation of the solutions, and escalating quality problems to higher management levels. This position has the authority to suspend unsatisfactory work and control further processing or installation of non-conforming materials. The authority to stop work delegated to Nuclear Oversight personnel is delineated in procedures. The director of nuclear oversight reports to the chief nuclear officer.

13.1.1.3.1.7 **Director of Nuclear Analysis and Fuel**

The director of nuclear analysis and fuel is responsible for providing nuclear fuel and related business and technical support consistent with the operational needs of the plant. The director of nuclear analysis and fuel is assisted by functional managers of fuel procurement, safety analysis, core design, probabilistic risk assessment, spent fuel storage and handling, fuel performance, and reactor engineering. The director of nuclear analysis and fuel reports to the vice president of nuclear engineering.

13.1.1.3.1.8 **Director of Nuclear Engineering, Corporate**

The director of nuclear engineering, corporate, is responsible for providing engineering support to the nuclear stations in the areas of design, projects, and programs, including the fire protection program. This position is supported by the functional managers responsible for design engineering, project engineering, and engineering programs.

13.1.1.3.1.9 **Fire Protection Engineer**

The fire protection engineer is responsible for:

- Fire Protection Program requirements, including consideration of potential hazards associated with postulated fires, knowledge of building layout, and system design.
- Post-fire shutdown capability.
- Design, maintenance, surveillance, and quality assurance of fire protection features (e.g., detection systems, suppression systems, barriers, dampers, doors, penetration seals, and fire brigade equipment.
- Fire prevention activities (administrative controls).
- Pre-fire planning, including review and updating of pre-fire plans at least every two years.

The fire protection engineer reports to the site vice president through the corporate director of nuclear engineering. Additionally, the fire protection engineer works with the operations department to coordinate activities and program requirements. In accordance with RG 1.1.89, the fire protection engineer is an individual who has been delegated authority commensurate with the responsibilities of the position, and who has available staff personnel knowledgeable in fire protection and nuclear safety.

13.1.1.3.2 **Site Organization (Operating)**

This position is supported by functional managers responsible for design engineering, project engineering, and engineering programs.

13.1.1.3.2.1 **Site Vice President**

The site vice president reports to the senior vice president of nuclear operations. The site vice president is directly responsible for management and direction of activities associated with the efficient, safe, and reliable operation of the nuclear station, except for those functions delegated to the vice president of nuclear engineering, the vice president of support services, and the director of nuclear oversight. The site vice president is assisted in management and technical support activities by the plant manager and the director in charge of nuclear safety and licensing. The site vice president is responsible for the site fire protection program through the fire protection engineer.

13.1.1.3.2.2 Site Director of Nuclear Engineering

The site director of nuclear engineering is the on-site lead position for engineering and reports to the vice president of nuclear engineering. The site director of nuclear engineering is responsible for engineering activities related to design engineering, system engineering, project engineering, program engineering, and component engineering. The site director of nuclear engineering directs functional managers responsible for each of these engineering areas.

13.1.1.3.2.2.1 Functional Managers in Charge of System Engineering

The functional managers in charge of system engineering supervise a technical staff of engineers and other engineering specialists and coordinate their work with that of other groups. System engineering staff includes reactor engineering. The functional manager in charge of system engineering reports to the site director of nuclear engineering and is responsible for providing direction and guidance to system engineers as follows:

- Monitoring the efficiency and proper operation of balance of plant and reactor systems.
- Planning programs for improving equipment performance, reliability, or work practices.
- Conducting operational tests and analyzing the results.
- Identification of plant spare parts for systems within his/her cognizance.

13.1.1.3.2.2.2 Functional Managers in Charge of Design Engineering

The functional managers in charge of design engineering report to the site director of nuclear engineering and are responsible for:

- Resolution of design issues.
- On-site development of design related change packages and plant modifications.
- Management of contractors who may perform modification or construction activities.
- Maintaining configuration control program.

13.1.1.3.2.2.3 Functional Managers in Charge of Engineering Programs

The functional managers in charge of engineering programs report to the site director of nuclear engineering and are responsible for programs such as:

- Materials engineering
- Performance/ISI engineering
- Valve engineering
- Maintenance rule tracking and trending
- Piping erosion corrosion
- In-service testing
- Fire protection
- Predictive Analysis

13.1.1.3.2.2.4 Functional Manager in Charge of Projects

The functional manager in charge of projects reports to the site director of nuclear engineering and is responsible for:

- Development of maintenance programs and specifications of selected plant equipment.
- Planned upgrades to equipment such as turbine rotors and major component replacement.
- Implementation of effective project management of contractors.
- Implementation of effective project management methods and procedures, including cost controls, for implementation of modifications and construction activities.

13.1.1.3.2.2.5 Functional Manager in Charge of Component Engineering

The functional manager in charge of component engineering reports to the site director of nuclear engineering. This position is supported by a staff of experts in specialized areas including pumps, AOVs, MOVs, and safety and relief valves. The staff provides support to the maintenance department and to other engineering groups.

13.1.1.3.2.3 Manager of Organizational Effectiveness

The responsibilities of the manager of organizational effectiveness include establishing processes and procedures to facilitate identification and correction of conditions adverse to quality and implementing corrective actions. The functional manager in charge of corrective actions and assessments reports to the director of nuclear safety and licensing.

13.1.1.3.2.4 Functional Manager in Charge of Plant Licensing

The functional manager in charge of plant licensing is responsible for providing a coordinated focus for interface with the NRC, and for technical direction and administrative guidance to the licensing staff for the following activities:

- Developing licensee event reports (LERs) and responding to notices of violations.
- Preparing/submitting license amendments and updating the FSAR.
- Tracking commitments and answering generic letters.
- Analyzing operating experience data and monitoring industry issues.
- Preparing the station for special NRC inspections, interfacing with NRC inspectors, and interpreting NRC regulations.
- Maintaining the licensing basis.

The functional manager in charge of plant licensing reports to the director of nuclear safety and licensing.

13.1.1.3.2.5 Functional Manager in Charge of Emergency Preparedness

The functional manager in charge of emergency preparedness is responsible for:

- Coordinating and implementing the plant emergency response plan with state and local emergency plans.
- Developing, planning, and executing emergency drills and exercises.

- Emergency action level development.
- NRC reporting associated with 10 CFR 50.54(q).

The functional manager in charge of emergency preparedness reports to the vice president of nuclear support services through the corporate emergency planning and support management.

13.1.1.3.2.6 **Manager of Nuclear Training**

The manager of nuclear training is responsible for training programs at the site required for the safe and proper operation and maintenance of the plant as described in Section 13.1.1.2.8. The manager of nuclear training supervises a staff of training supervisors who coordinate the development, preparation, and presentation of training programs for nuclear plant personnel and reports through corporate-training and development to the vice president of nuclear support services.

13.1.1.3.2.7 **Functional Manager of Supply Chain Services**

The functional manager of supply chain services is responsible for providing sufficient and proper materials to support the material needs of the plant and performing related activities including:

- Procedure development
- Materials storage
- Supply system database maintenance
- Meeting quality assurance and internal audit requirements

The functional manager of supply chain services is also responsible for site purchasing. This position reports to the vice president of nuclear support services through the corporate supply chain organization.

13.1.1.3.2.8 **Manager of Nuclear Protection**

The manager of nuclear protection is responsible for:

- Implementation and enforcement of security directives, procedures, and instructions received from appropriate authorities.
- Day-to-day supervision of the security guard force.
- Administration of the security program.
- Training the security force.
- Implementing the fitness-for-duty program.

The manager of nuclear protection reports to the vice president of support services via corporate security management.

13.1.1.3.2.9 Manager of Nuclear Oversight

The manager of nuclear oversight is responsible for those functions listed in Section 13.1.1.2.7. The manager of nuclear oversight reports to corporate oversight management.

13.1.1.4 Qualifications of Technical Support Personnel

Personnel of the technical support organization meet the education and experience qualifications for those described in ANSI/ANS-3.1 (Reference 13.1-201) as endorsed and amended by RG 1.8.

13.1.2 Operating Organization

13.1.2.1 Plant Organization

The plant management, technical support, and plant operating organizations are shown in Figure 13.1-204. The operating organization is described in Sections 13.1.1.3 and 13.1.2. The on-shift organization is shown in Figure 13.1-203. Additional personnel are required to augment normal staff during outages.

Nuclear plant employees are responsible for reporting problems with plant equipment and facilities. They are required to identify and document equipment problems in accordance with the QA program. QA program requirements as they apply to the operating organization are described in Section 17.5.

Rules of practice are met through administrative controls as described in Section 17.5. These controls include:

- Establishment of a quality assurance program for the operational phase
- Preparation of procedures necessary to carry out an effective quality assurance program
- A program for review and audit of activities affecting plant safety
- Programs and procedures for rules of practice

Managers and supervisors within the plant operating organization are responsible for establishing goals and expectations for their organization and to reinforce behaviors that promote radiation protection. Specifically, managers and supervisors are responsible for the following, as applicable to their position within the plant organization:

- Interfacing directly with radiation protection staff to integrate radiation protection measures into plant procedures and designing documents into the planning, scheduling, conduct, and assessment of operations and work.
- Notifying radiation protection personnel promptly when radiation protection problems occur or are identified, taking corrective actions, and resolve deficiencies associated with operations, procedures, systems, equipment, and work practices.
- Training site personnel on radiation protection and providing periodic retraining in accordance with 10 CFR 19 so that personnel are properly instructed and briefed for entry into restricted areas.
- Periodically observing and correcting, as necessary, radiation worker practices.
- Supporting radiation protection management in implementing the radiation protection program.
- Maintaining exposures to site personnel ALARA.

13.1.2.1.1 **Site Vice President**

The site vice president reports to the senior vice president of nuclear operations. The site vice president is directly responsible for management and direction of activities associated with the efficient, safe, and reliable operation of the nuclear station, except for those functions delegated to the vice president of nuclear engineering, the vice president of nuclear support services, and the director of nuclear oversight. The site vice president is assisted in management and technical support activities by the plant manager and the plant safety and licensing (S&L) director. Executive management establishes expectations such that a high level of quality, safety, and efficiency is achieved in aspects of plant operations and support activities through an effective management control system and an organization selected and trained to meet the above objectives.

Additionally, the site vice president has overall responsibility for occupational and public radiation safety. Radiation protection responsibilities of the site vice president are consistent with the guidance in RG 8.8 and RG 8.10, including the following:

- Providing management radiation protection policy throughout the plant organization
- Providing an overall commitment to radiation protection by the plant organization
- Interacting with and supporting the manager in charge of radiation protection on implementation of the radiation protection program
- Supporting identification and implementation of cost-effective modifications to plant equipment, facilities, procedures and processes to improve radiation protection controls and reduce exposures
- Establishing plant goals and objectives for radiation protection
- Maintaining exposures to site personnel ALARA
- Supporting timely identification, analysis, and resolution of radiation protection problems (e.g., through the plant corrective action program)
- Providing training to site personnel on radiation protection in accordance with 10 CFR 19
- Establishing an ALARA Committee with delegated authority from the site that includes the managers in charge of operations, maintenance, engineering, and radiation protection to help provide for effective implementation of line organization responsibilities for maintaining worker doses ALARA

The site vice president is responsible for the site fire protection program through the fire protection engineer.

The succession of responsibility for overall plant instructions or special orders in the event of absences, incapacitation of personnel, or other emergencies is as follows, unless otherwise designated in writing:

1. The site vice president
2. The plant manager
3. The manager of nuclear operations

The succession of authority includes the authority to issue standing or special orders as required.

13.1.2.1.1.1 **Plant Manager**

The plant manager reports to the site vice president, is responsible for safe operation of the plant, and has control over onsite activities necessary for safe operation and maintenance of the plant including the following:

- Operations
- Maintenance and modification
- Outage and planning management

13.1.2.1.1.2 **Director of Nuclear Safety & Licensing**

The director of nuclear safety and licensing reports to the site vice president, is responsible for safe operation of the plant, and has control over onsite activities necessary for safe operation and maintenance of the plant including the following:

- Procedures
- Licensing
- Radiation protection
- Chemistry and radiochemistry
- Organizational effectiveness

13.1.2.1.1.3 **Maintenance Manager**

Maintenance of the plant is performed by the maintenance department mechanical, electrical, and instrumentation and control disciplines. The functions of this department are to perform preventive and corrective maintenance, equipment testing, and to implement modifications as necessary.

The manager in charge of plant maintenance is responsible for the performance of preventive and corrective maintenance and modification activities required to support operations, including compliance with applicable standards, codes, specifications, and procedures. The maintenance manager is responsible for the development of maintenance programs. The maintenance manager reports to the plant manager and provides direction and guidance to the maintenance discipline functional managers and maintenance support staff.

13.1.2.1.1.4 Maintenance Discipline Functional Managers

The functional managers of each maintenance discipline (mechanical, electrical, instrumentation and control, and support) are responsible for maintenance activities within their discipline including plant modifications. They provide guidance in maintenance planning and craft supervision. They establish the necessary manpower levels and equipment requirements to perform both routine and emergency type maintenance activities, seeking the services of others in performing work beyond the capabilities of the plant maintenance group. Each discipline functional manager is responsible for liaison with other plant staff organizations to facilitate safe operation of the station. These functional managers report to the maintenance manager.

13.1.2.1.1.5 Maintenance Discipline Supervisors

The maintenance discipline supervisors and assistant supervisors (mechanical, electrical, and instrumentation and control) supervise maintenance activities, assist in the planning of future maintenance efforts, and guide the efforts of the craft within their discipline. The maintenance discipline supervisors report to the appropriate maintenance discipline functional managers.

13.1.2.1.1.6 Maintenance Mechanics, Electricians, and Instrumentation and Control Technicians

The discipline craft perform electrical and mechanical maintenance and I&C tasks as assigned by the discipline supervisors. They troubleshoot, inspect, repair, maintain, and modify plant equipment and perform Technical Specification surveillances on equipment for which they have cognizance. They perform these tasks in accordance with approved procedures and work packages.

13.1.2.1.1.7 Outage and Planning Manager

The outage and planning manager is responsible for the support functions described in Section 13.1.1.2.5. This manager safely fulfills the responsibilities of planning and scheduling all plant work through a staff which includes a functional manager in each area of planning, scheduling, and outages. The outage and planning manager reports to the plant manager.

13.1.2.1.1.8 **Manager of Radiation Protection and Chemistry**

The manager radiation protection and chemistry has the direct responsibility for providing adequate protection of the health and safety of personnel working at the plant and members of the public during activities covered within the scope and extent of the license. This manager's radiation protection responsibilities are consistent with the guidance in RG 8.8 and RG 8.10. They include:

- Managing the radiation protection organization
- Establishing, implementing, and enforcing the radiation protection program
- Providing radiation protection input to facility design and work planning
- Tracking and analyzing trends in radiation work performance and taking necessary actions to correct adverse trends
- Supporting the plant emergency preparedness program and assigning emergency duties and responsibilities within the radiation protection organization
- Delegating authority to appropriate radiation protection staff to stop work or order an area evacuated (in accordance with approved procedures) when, in his or her judgment, the radiation conditions warrant such an action and such actions are consistent with plant safety
- Managing the radioactive waste programs

The manager radiation protection and chemistry reports to the director of safety and licensing and is assisted by the supervisors in charge of radiation protection and the functional manager in charge of chemistry.

13.1.2.1.1.9 **Supervisors in Charge of Radiation Protection**

The supervisors in charge of radiation protection are responsible for carrying out the day-to-day operations and programs of the radiation protection department as listed in Section 13.1.1.2.3, to promote safe and efficient plant operation.

Supervisors in charge of radiation protection report to the manager of radiation protection and chemistry.

13.1.2.1.1.10 **Radiation Protection Technicians**

Radiation protection technicians (RPTs) directly carry out responsibilities defined in the radiation protection program and procedures. In accordance with Technical Specifications, an RPT is on site whenever there is fuel in the vessel.

The following are some of the duties and responsibilities of the RPTs:

- In accordance with authority delegated by the manager in charge of radiation protection, stop work or order an area evacuated (in accordance with approved procedures) when, in his or her judgment, the radiation conditions warrant such an action and such actions are consistent with plant safety
- Provide coverage and monitor radiation conditions for jobs potentially involving significant radiation exposure
- Conduct surveys, assess radiation conditions, and establish radiation protection requirements for access to and work within restricted, radiation, high radiation, very high radiation, airborne radioactivity areas, and areas containing radioactive materials
- Provide control over the receipt, storage, movement, use, and shipment of licensed radioactive materials
- Review work packages, proposed design modifications, and operations and maintenance procedures to facilitate integration of adequate radiation protection controls and dose-reduction measures
- Review and oversee implementation of plans for the use of process or other engineering controls to limit the concentrations of radioactive materials in the air
- Provide personnel monitoring and bioassay services
- Maintain, prescribe, and oversee the use of respiratory protection equipment
- Perform assigned emergency response duties.

13.1.2.1.1.11 **Functional Manager in Charge of Chemistry**

The functional manager in charge of chemistry is responsible for development, implementation, and direction and coordination of the chemistry, radiochemistry, and non-radiological environmental monitoring programs. This area includes overall operation of the hot lab, cold lab, emergency offsite facility lab, and non-radiological environmental monitoring. The functional manager in charge of chemistry is responsible for the development, administration, and implementation of procedures and programs which provide for effective compliance with environmental regulations. The functional manager in charge of chemistry reports to the director of safety and licensing via the manager of radiation protection and chemistry and directly supervises the chemistry supervisors.

The functional manager in charge of chemistry is responsible for assuring that a chemistry technician is on site whenever the unit is in modes other than cold shutdown or refueling.

13.1.2.1.1.12 **[Deleted]**

13.1.2.1.2 Operations Department

All operations activities are conducted with safety of personnel, the public, and equipment as the overriding priority. Management personnel of the operations department are responsible for:

- Operation of station equipment
- Monitoring and surveillance of safety- and non-safety-related equipment
- Fuel loading
- Providing the nucleus of emergency and fire-fighting teams

The operations department maintains sufficient licensed and senior licensed operators to staff the MCR continuously using a crew rotation system. The operations department is under the authority of the manager in charge of operations who, through the supervisor in charge of shift operations, directs the day-to-day operation of the plant.

Specific duties, functions, and responsibilities of key shift members are discussed in Section 13.1.2.1.2.5 through Section 13.1.2.1.2.9 and in plant administrative procedures and the Technical Specifications. The minimum shift manning requirements are shown in Table 13.1-202. Expected staffing levels are provided in Table 13.1-201.

For activities that do not require an operator's license, resources of the operations organization may be shared between units. These activities may include administrative functions and tagging. To operate or supervise the operation of more than one unit, an operator (SRO or RO) must hold an appropriate, current license for each unit. A single management organization oversees the operations group for the station units.

The Operations Support Section is staffed with sufficient personnel to provide support activities for the operating shifts and overall operations department. The following is an overview of the operations organization.

13.1.2.1.2.1 **Manager of Nuclear Operations**

The manager of nuclear operations has overall responsibility for the day-to-day operation of the plant. The manager of nuclear operations reports to the plant manager and is assisted by the functional managers of nuclear shift operations, operations support, and operations maintenance support. Either the manager of nuclear operations or the functional manager of nuclear shift operations is SRO licensed.

13.1.2.1.2.2 **Functional Manager of Nuclear Shift Operations**

The functional manager of nuclear shift operations, under the direction of the manager of nuclear operations, is responsible for:

- Shift plant operations in accordance with the operating license, Technical Specifications, and written procedures
- Providing supervision of operating shift personnel for operational shift activities including those of emergency and firefighting teams
- Coordinating with the functional manager of operations support and other plant staff sections
- Verifying that nuclear plant operating records and logs are properly prepared, reviewed, evaluated and turned over to the functional manager of operations support

The functional manager of nuclear shift operations is assisted in these areas by the on-shift operations manager who directs the operating shift personnel. The functional manager of nuclear shift operations may assume the duties of the manager of nuclear operations in the event of an absence.

13.1.2.1.2.3 **Functional Manager of Operations Support**

The functional manager of operations support, under the direction of the manager of nuclear operations is responsible for:

- Directing and guiding plant operations support activities in accordance with the operating license, Technical Specifications, and written procedures
- Providing supervision of operating support personnel and operations support activities, and coordination of support activities

- Providing for nuclear plant operating records and logs to be turned over to the nuclear records group for maintenance as quality records
- Supervising operating procedure maintenance

The supervisor of operations support is assisted by the supervisors of work management, radwaste operations, operations procedures group, and other support personnel. In the absence of the operations manager, the supervisor of operations support may assume the duties and responsibilities of this position.

13.1.2.1.2.4 **Functional Manager of Operations Maintenance Support**

The functional manager of nuclear operations maintenance support is a licensed SRO reporting directly to the manager of nuclear operations. Responsibilities of this position include:

- Valve lineups for maintenance and testing activities.
- Equipment tagging
- Review and authorization of maintenance, surveillance, or other work or testing.
- Keeping the operations shift manager and other operations personnel informed of activities for which they need to be cognizant.
- Verifying that work and testing is safe and appropriate for the existing conditions of the plant.
- Tracking the work and testing to provide assurance that any LCOs or other requirements will not be exceeded.

13.1.2.1.2.5 **Operations Shift Manager**

The operations shift manager is a licensed senior reactor operator (SRO) responsible for the control room command function, and is the plant manager’s direct management representative for the conduct of operations. The operations shift manager has the responsibility and authority to direct the activities and personnel onsite as required to:

- Protect the health and safety of the public, the environment, and personnel on the plant site
- Prevent damage to site equipment and structures
- Comply with the operating license

The operations shift manager retains this responsibility and authority until formally relieved of operating responsibilities by a licensed SRO. Additional responsibilities of the operations shift manager include:

- Directing nuclear plant employees to report to the plant for response to potential and real emergencies
- Seeking the advice and guidance of the shift technical advisor and others in executing his duties whenever in doubt as to the proper course of action
- Promptly informing responsible supervisors of significant actions affecting their responsibilities
- Participating in operator training, retraining, and requalification activities from the standpoint of providing guidance, direction, and instruction to shift personnel

The operations shift manager is assisted in carrying out the above duties by the on-shift unit supervisors and the operating shift personnel. As shown on Figure 13.1-203, the shift operations manager reports to the functional manager of nuclear shift operations.

13.1.2.1.2.6 **On-Shift Unit Supervisor**

The on-shift unit supervisor is a licensed SRO. The main functions of the on-shift unit supervisor are to administratively support the operations shift manager such that the "command function" is not overburdened with administrative duties and to supervise the licensed and non-licensed operators in carrying out the activities directed by the operations shift manager. Other duties and responsibilities include:

- Being aware of maintenance and testing performed during the shift
- Directing reactor shutdown if conditions warrant this action
- Informing the operations shift manager and other station management in a timely manner of conditions which may affect public safety, plant personnel safety, plant capacity or reliability, or cause a hazard to equipment
- Initiating immediate corrective action as directed by the operations shift manager in any upset situation until assistance, if required, arrives
- Participating in operator training, retraining, and requalification activities from the standpoint of providing guidance, direction, and instruction to shift personnel

- Responding conservatively to instrument indications unless they are proved to be incorrect
- Adhering to the plant's technical specifications
- Reviewing routine operating data to assure safe operation

As shown on Figure 13.1-203, the on-shift unit supervisor reports directly to the operations shift manager.

13.1.2.1.2.7 **Reactor Operator**

Reactor operators (RO) are licensed personnel and normally report to the on-shift unit supervisor. They are responsible for routine plant operations and performance of major evolutions at the direction of the on-shift unit supervisor. The RO duties and responsibilities include:

- Monitoring control room instrumentation
- Responding to plant or equipment abnormalities in accordance with approved plant procedures
- Directing the activities of non-licensed operators
- Documenting operational activities, plant events, and plant data in shift logs
- Responding conservatively to instrument indications unless they are proved to be incorrect
- Adhering to the plant's technical specifications
- Reviewing routine operating data to assure safe operation
- Initiating plant shutdowns or scrams or other compensatory actions when:
  - Observation of plant conditions indicates a nuclear safety hazard exists
  - Approved procedures so direct
  - The operator determines that the safety of the reactor is in jeopardy
  - Operating parameters exceed any of the reactor protection system setpoints and automatic shutdown does not occur

Whenever there is fuel in the reactor vessel, at least one reactor operator is in the control room monitoring the status of the unit at the main control panel. The RO assigned to the main control panel is designated the Operator-At-The Controls (OATC) and conducts monitoring and operating activities in accordance with the guidance set forth in RG 1.114, which is further described in Section 13.1.2.1.3.

13.1.2.1.2.8 **Non-Licensed Operator**

The non-licensed operators perform routine duties outside the control room as necessary for continuous, safe plant operation including:

- Assisting in plant startup, shutdown, surveillance, and emergency response by manually or remotely changing equipment operating conditions, placing equipment in service, or securing equipment from service at the direction of the RO
- Performing assigned tasks in procedures and checklists such as valve manipulations for plant startup or data sheets on routine equipment checks, and making accurate entries according to the applicable procedure, data sheet, or checklist
- Assisting in training of new employees and improving and upgrading their own performance by participating in the applicable sections of the training program

13.1.2.1.2.9 **Shift Technical Advisor**

The station is committed to meeting NUREG-0737 TMI Action Plan item I.A.1.1 for shift technical advisors (STAs). The STA reports directly to the shift manager and provides advanced technical assistance to the operating shift complement during normal and abnormal operating conditions. The STA's responsibilities are detailed in plant administrative procedures as required by TMI Action Plan I.A.1.1 and NUREG-0737, Appendix C. These responsibilities include:

- Monitoring core power distribution and critical parameters
- Assisting the operating shift with technical expertise during normal and emergency conditions
- Evaluating technical specifications, special reports, and procedural issues

The STA contributes to operations safety by independently observing plant status and advising shift supervision of conditions that could compromise plant safety. During transients or accident situations, the STA independently assesses plant conditions and provides technical assistance and advice to mitigate the incident and minimize the effect on personnel, the environment, and plant equipment.

An SRO on shift who meets the qualifications for the combined SRO/STA position specified for Option 1 of Generic Letter 86-04 (Reference 13.1-202) may also serve as the STA. If this option is used for a shift, the separate STA position may be eliminated for that shift.

13.1.2.1.3 **Conduct of Operations**

Station operations are controlled and coordinated through the control room. Maintenance activities, surveillances, and removal from/return to service of SSCs affecting the operation of the plant may not commence without the authority of senior control room personnel. The rules of practice for control room activities, as described by administrative procedures, which are based on RG 1.114, address the following:

- Position/placement of the workstation for the operator at the controls and the expected area of the control room where the supervisor/manager in charge on shift should spend the majority of on-shift time
- Definition and outline of “surveillance area” and requirement for continuous surveillance by the operator at the controls
- Relief requirements for operator at the controls and the supervisor/manager in charge on shift

In accordance with 10 CFR 50.54 (i), (j), (k), (l), and (m):

- Reactivity controls may be manipulated only by licensed operators and senior operators except as allowed for training under 10 CFR 55
- Apparatus and mechanisms other than controls which may affect reactivity or power level of the reactor shall be operated only with the consent of the operator at the controls or the manager/supervisor in charge on-shift
- An operator or senior operator shall be present at the controls at all times during the operation of the facility
- For each shift, operations management designates one or more SROs to be responsible for directing the licensed activities of licensed operators
- An SRO shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial start-up and approach to power, recovery from an unplanned or unscheduled shut-down or significant reduction in power, and refueling, or as otherwise prescribed in the facility license

- Minimum shift staffing for operations personnel is shown in Table 13.1-202
- With the unit in modes other than cold shutdown or refueling, there shall be one SRO in the control room at all times. In addition, there shall be one RO or one SRO at the controls whenever there is fuel in the reactor vessel

13.1.2.1.4 **Operating Shift Crews**

Plant administrative procedures implement the required shift staffing. These provisions establish crews with sufficient qualified plant personnel to staff the operational shifts and be readily available in the event of an abnormal or emergency situation. The objective is to operate the plant with the required staff and to develop work schedules that minimize overtime for plant staff members who perform safety-related functions. Work hour limitations and shift manning requirements defined by TMI Action Plan I.A.1.3 are addressed in station procedures. Shift crew staffing plans may be modified during refueling outages to accommodate safe and efficient completion of outage work in accordance with work hour limitations established in administrative procedures.

The minimum composition of an operating shift depends on the operational mode, as shown in Table 13.1-202. Reporting relationships for these positions are shown in Figure 13.1-203.

NAPS COL 9.5.1-10-H

13.1.2.1.5 **Fire Brigade**

The plant is designed, and the fire brigade organized, to be self-sufficient with respect to fire fighting activities. The fire brigade is organized to deal with fires and related emergencies that could occur. It consists of a fire brigade leader and a sufficient number of team members to be consistent with the equipment that must be put in service during a fire emergency. A sufficient number of trained and physically qualified fire brigade members are available on site during each shift. The fire brigade consists of at least five members on each shift. Members of the fire brigade are knowledgeable of building layout and system design. The assigned fire brigade members for any shift do not include the operations shift manager nor any other members of the minimum shift operating crew necessary for safe shutdown of the unit, nor do they include any other personnel required for other essential functions during a fire emergency. Fire brigade members for a shift are designated in accordance with established procedures at the beginning of the shift. The fire brigade for Unit 3 does not include personnel assigned to Units 1 and 2.

The brigade leader and at least two brigade members have sufficient training in, or knowledge of, plant systems to understand the effects of fire and fire suppressents on safe-shutdown capability. The brigade leader has training or experience necessary to assess the potential safety consequences of a fire and advise control room personnel, as evidenced by possession of an operator's license or equivalent knowledge of plant systems. The qualification of fire brigade members includes an annual physical examination to determine their ability to perform strenuous firefighting activities.

13.1.3 Qualification Requirements of Nuclear Plant Personnel

13.1.3.1 Minimum Qualification Requirements

Qualifications of managers, supervisors, operators, and technicians of the operating organization meet the requirements for education and experience described in ANSI/ANS-3.1 (Reference 13.1-201), as endorsed and amended by RG 1.8. For operators and SROs, these requirements are modified in Section 13.2.

13.1.3.2 Qualification Documentation

Resumes and other documentation of qualification and experience of initial appointees to appropriate management and supervisory positions are available for review by regulators upon request after position vacancies are filled.

13.1.4 COL Information

13.1-1-A Organizational Structure

NAPS COL 13.1-1-A This COL item is addressed in Sections 9.5.1.15.3, 13.1.1 through 13.1.3.

13.1.5 References

13.1-201 American Nuclear Society, "American National Standard for Selection, Qualification, and Training of Personnel for Nuclear Power Plant," ANSI/ANS -3.1.

13.1-202 U.S. Nuclear Regulatory Commission, "Generic Letter 86-04, Policy Letter, Engineering Expertise on Shift."

NAPS COL 13.1-1-A

Table 13.1-201 Generic Position/Site Specific Position Cross Reference

Nuclear Function	Function Position (ANS-3.1-1993 section)	Nuclear Plant Position (Site-Specific)	Estimated Numbers of Full Time Equivalents* Design Review Phase	Construction Phase	Pre-op Phase	Operational Phase
Executive management	chief nuclear officer (n/a)	CNO Dominion	1**	1**	1**	1**
	senior executive, nuclear operations (n/a)	Senior Vice President, Nuclear Operations	1**	1**	1**	1**
	site executive	Site Vice President - NAPS	1***	1***	1***	1***
Nuclear support	executive, operations support (n/a)	Vice President - Nuclear Support Services	1**	1**	1**	1**
	executive, construction (n/a)	Vice President - Nuclear Development	1**	1**	1**	
	executive, engineering and technical services (n/a)	Vice President - Engineering	1**	1**	1**	1**
Plant management	plant manager (4.2.1)	Plant Manager			1	1
	safety and licensing manager (n/a)	Director Nuclear Station Safety & Licensing			1	1

NAPS COL 13.1-1-A

Table 13.1-201 Generic Position/Site Specific Position Cross Reference

Nuclear Function	Function Position (ANS-3.1-1993 section)		Nuclear Plant Position (Site-Specific)	Estimated Numbers of Full Time Equivalents* Design Review Phase	Construction Phase	Pre-op Phase	Operational Phase
Operations	manager	(4.2.2)	Manager, Operations			1	1
operations, plant	functional manager	(4.3.8)	Operations – Shift Supervisor			1	1
operations, admin	functional manager	(4.3.8)	Operations – Support Supervisor			1	1
operations, (on-shift)	functional manager	(4.4.1)	Shift Manager			6	6
	supervisor	(4.4.2)	Unit Supervisor			5	5
	supervisor	(4.4.2)	Supervisor, Work Control			5	5
	supervisor	(4.6.2)	STA****			5	5
	licensed operator	(4.5.1)	Control Room Operator			15	24
	non-licensed operator	(4.5.2)	Non-licensed Operator		6	24	30
	rad waste operator	(4.5.2)	Rad Waste Operator			1	2
Engineering	manager	(4.2.4)	Director, Nuclear Engineering	1	1	1	1
projects	functional manager	(4.3.9)	Manager, Projects		1	1	1
	projects engineer	(n/a)	Project Engineer	1	3	3	5
system engineering	functional manager	(4.3.9)	Supervisor, System Engineering		1	4	4
	system engineer	(4.6.1)	System Engineer	1	4	16	16

NAPS COL 13.1-1-A

Table 13.1-201 Generic Position/Site Specific Position Cross Reference

Nuclear Function	Function Position (ANS-3.1-1993 section)	Nuclear Plant Position (Site-Specific)	Estimated Numbers of Full Time Equivalents*: Design Review Phase	Construction Phase	Pre-op Phase	Operational Phase
design engineering	functional manager (4.3.9)	Supervisor, Design Engineering	1	1	1	1
	design engineer (4.6 – staff engineer)	Design Engineer	3	5	10	15
safety and engineering analysis	functional manager (4.3.9)	Manager, Nuclear Safety Engineering		1	1	1
	analysis engineer (4.6–staff engineer)	Analysis Engineer		1	1	1
engineering programs	functional manager (4.3.9)	Supervisor, Engineering Programs		1	1	1
	programs engineer (4.6–staff engineer)	Programs Engineer		6	12	12
reactor engineering	functional manager (4.3.9)	Supervisor, Reactor Engineering			1	1
	reactor engineer (4.6–staff engineer)	Reactor Engineer		1	3	3
Chemistry	functional manager (4.3.2)	Manager, Radiation Protection & Chemistry		1***	1***	1***
	supervisor (4.4.5)	Chemistry Supervisor		1	1	2
	technician (4.5.3.1)	Chemistry Technician		2	6	10

NAPS COL 13.1-1-A

Table 13.1-201 Generic Position/Site Specific Position Cross Reference

Nuclear Function	Function Position (ANS-3.1-1993 section)		Nuclear Plant Position (Site-Specific)	Estimated Numbers of Full Time Equivalents*: Design Review Phase	Construction Phase	Pre-op Phase	Operational Phase
Radiation Protection	functional manager	(4.3.3)	Manager, Radiation Protection & Chemistry		1***	1***	1***
	supervisor	(4.4.6)	Health Physics Supervisor		2	6	8
	technician	(4.5.3.2)	Health Physics Technician		4	12	18
	ALARA specialist	(n/a)	ALARA Specialist		1	3	3
	decon technician	(n/a)	Decon Technician		2	6	6
Maintenance	manager	(4.2.3)	Manager, Maintenance			1	1
instrumentation and control	supervisor	(4.4.7)	Supervisor, Instrumentation and Control		1	1	1
	supervisor	(4.4.7)	Assistant Supervisor, Instrumentation and Control		2	2	2
	technician	(4.5.3.3)	Instrumentation and Control Technician		4	20	30
mechanical	supervisor	(4.4.9)	Supervisor, Mechanical		1	1	1
	supervisor	(4.4.9)	Assistant Supervisor, Mechanical		2	2	2
	technician	(4.5.7.2)	Mechanic		4	20	30
electrical	supervisor	(4.4.8)	Supervisor, Electrical		1	1	1
	supervisor	(4.4.8)	Assistant Supervisor, Electrical		2	2	2
	technician	(4.5.7.1)	Electrician		4	20	30

NAPS COL 13.1-1-A

Table 13.1-201 Generic Position/Site Specific Position Cross Reference

Nuclear Function	Function Position (ANS-3.1-1993 section)	Nuclear Plant Position (Site-Specific)	Estimated Numbers of Full Time Equivalents* Design Review Phase	Construction Phase	Pre-op Phase	Operational Phase
Planning and scheduling and outage	manager (4.2)	Manager, Outage & Planning			1***	1***
	functional manager (4.3)	Supervisor, Outage & Planning			1	1
	functional manager (4.3)	Supervisor, Scheduling			1	1
	functional manager (4.3)	Supervisor, Planning		1	1	1
Purchasing, and contracts	functional manager (4.3)	Manager, Supply Chain Services		1***	1***	1***
	procurement engineer (n/a)	Procurement Engineer		1	2	2
Quality assurance	functional manager (QAPD)	Director, Nuclear Oversight	1***	1***	1***	1***
	QA lead auditor (QAPD)	QA Auditor	1	1	1	1
	QA internal auditor (QAPD)	QA Auditor		2	2	8***
	QC inspector (QAPD)	QC Inspector		6	6	4***
	supplier auditor (QAPD)	Nuclear Quality Inspector		2	2	1***
	vendor surveillance QC inspector (QAPD)	Vendor Surveillance QC Inspector	2	6	4	4***
	nuclear fuel inspector (QAPD)	Nuclear Fuel Inspector		3***	3***	3***

Table 13.1-201 Generic Position/Site Specific Position Cross Reference

Nuclear Function	Function Position (ANS-3.1-1993 section)	Nuclear Plant Position (Site-Specific)	Estimated Numbers of Full Time Equivalents* Design Review Phase	Construction Phase	Pre-op Phase	Operational Phase
Training	functional manager (4.3.1)	Manager, Training		1***	1***	1***
	supervisor operations training (4.4.4)	Supervisor, Operations Training		1	1	1
	supervisor, simulator (4.4.4)	Supervisor, Simulator & Training Support		1	1	1
	operations training instructor (4.5.4)	Operations Training Instructor		10	10	10
	supervisor tech staff training (4.4.4)	Supervisor, Tech Training		1	1	1
	supervisor maintenance training (4.4.4)	Supervisor, Maintenance Training		1	1	1
	tech staff/maintenance instructors (4.5.4)	Tech Staff/Maintenance Instructor		7	7	7
Nuclear safety assurance	manager (4.2)	Director, Nuclear Safety & Licensing		1***	1***	1***
licensing	functional manager (4.3)	Supervisor, Licensing	1	1	1	1
	licensing engineer (n/a)	Licensing Engineer	4	4	4	2
corrective action	functional manager (4.3)	Supervisor, Station Nuclear Safety		1***	1***	1***
	corrective action engineer	Station Nuclear Safety Engineer		1	1	1

NAPS COL 13.1-1-A

Table 13.1-201 Generic Position/Site Specific Position Cross Reference

Nuclear Function	Function Position (ANS-3.1-1993 section)		Nuclear Plant Position (Site-Specific)	Estimated Numbers of Full Time Equivalents*: Design Review Phase	Construction Phase	Pre-op Phase	Operational Phase
Nuclear Protection Services							
emergency preparedness	functional manager	(4.3)	Manager, Emergency Planning		1**	1**	1**
	EP planner	(n/a)	EP Specialist		2***	2***	2***
security	functional manager	(4.3)	Manager, Security		1***	1***	1***
	first line supervisor	(4.4)	Supervisor, Nuclear Security		10***	10***	10***
	security officer	(n/a)	Security Officer		100***	100***	100***
Startup testing	supervisor	(4.4.12)	Startup Testing Supervisor		1	3	1
	startup test engineer		Startup Test Engineer		4	10	4
	supervisor	(4.4.11)	Preop Testing Supervisor		2	2	-
	preop test engineer	(n/a)	Preop Test Engineer		8	8	-

* Unless otherwise noted, the number in each block represents the estimated number of full time equivalents dedicated to the project.

** The number in this block indicates total positions in the nuclear organization.

*** Shared position with other North Anna units.

**** A senior reactor operator on shift who meets the qualifications for the combined SRO/STA position specified for Option 1 of Generic Letter 86-04 (Reference 13.1-202) may also serve as the STA. If this option is used for a shift, the separate STA position may be eliminated for that shift.

NAPS COL 13.1-1-A

Table 13.1-202 Minimum Shift Staffing for Unit 3

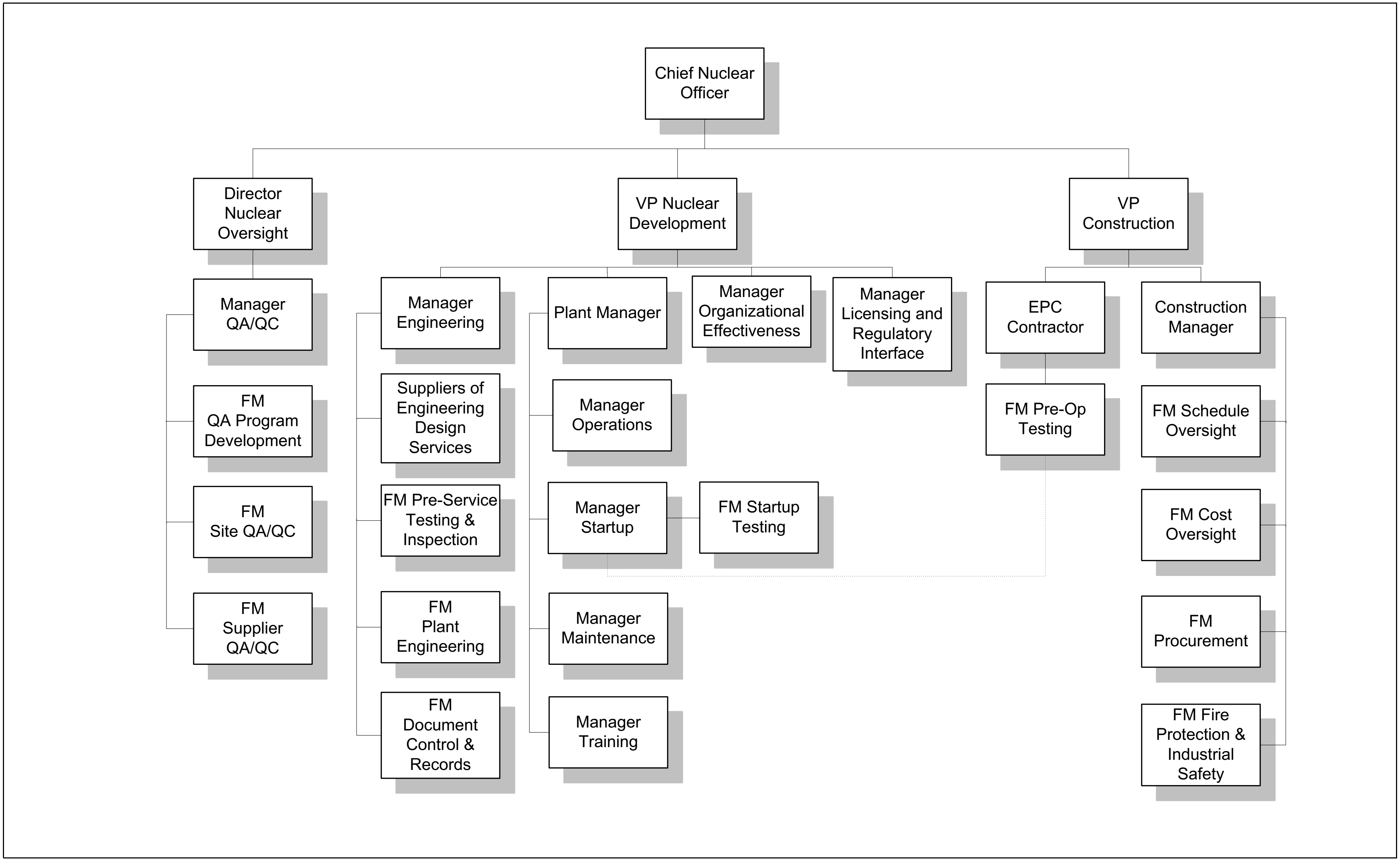
Unit Shutdown	1 SM (SRO) 1 RO 1 NLO
Unit Operating*	1 SM (SRO) 1 SRO 2 RO 2 NLO
SM – shift manager SRO – Licensed Senior Reactor Operator	RO – Licensed Reactor Operator NLO – non-licensed operator

Notes:

1) In addition, one Shift Technical Advisor (STA) is assigned during plant operation in modes other than cold shutdown or refueling. A shift manager or another SRO on shift, who meets the qualifications for the combined Senior Reactor Operator/Shift Technical Advisor (SRO/STA) position, as specified for option 1 of Generic Letter 86-04 (Reference 13.1-202), the commission's policy statement on engineering expertise on shift, may also serve as the STA. If this option is used for a shift, then the separate STA position may be eliminated for that shift.
2) In addition to the minimum shift organization above, during refueling a licensed senior reactor operator or senior reactor operator limited (fuel handling only) is required to directly supervise any core alteration activity.
3) A shift manager/supervisor (licensed SRO), is on site at all times when fuel is in the reactor.
4) A health physics technician is on site at all times where there is fuel in the reactor.
5) A chemistry technician is on site during plant operation in modes other than cold shutdown or refueling.

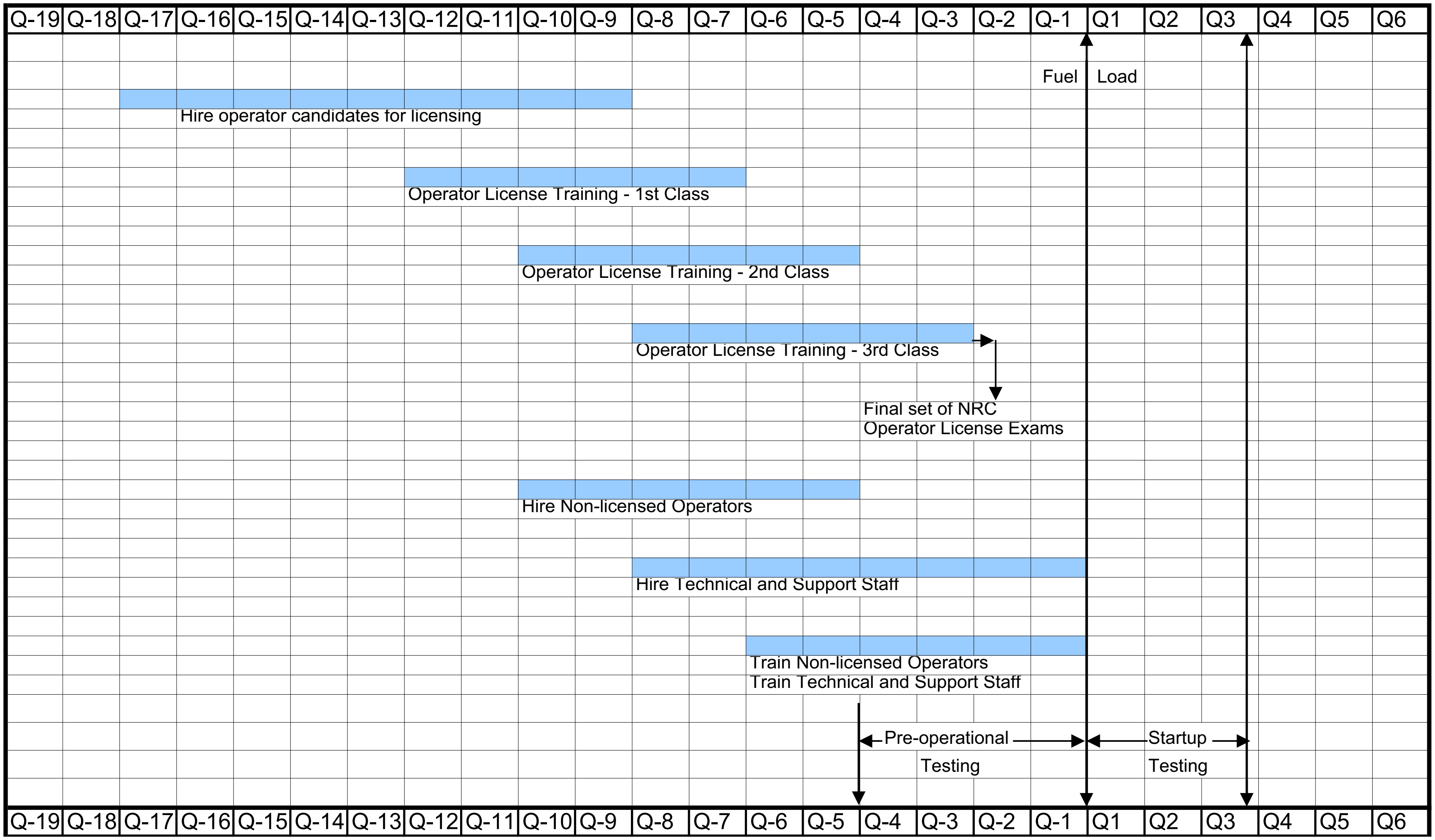
* Operating modes other than cold shutdown or refueling.

NAPS COL 13.1-1-A **Figure 13.1-201 Construction Organization**

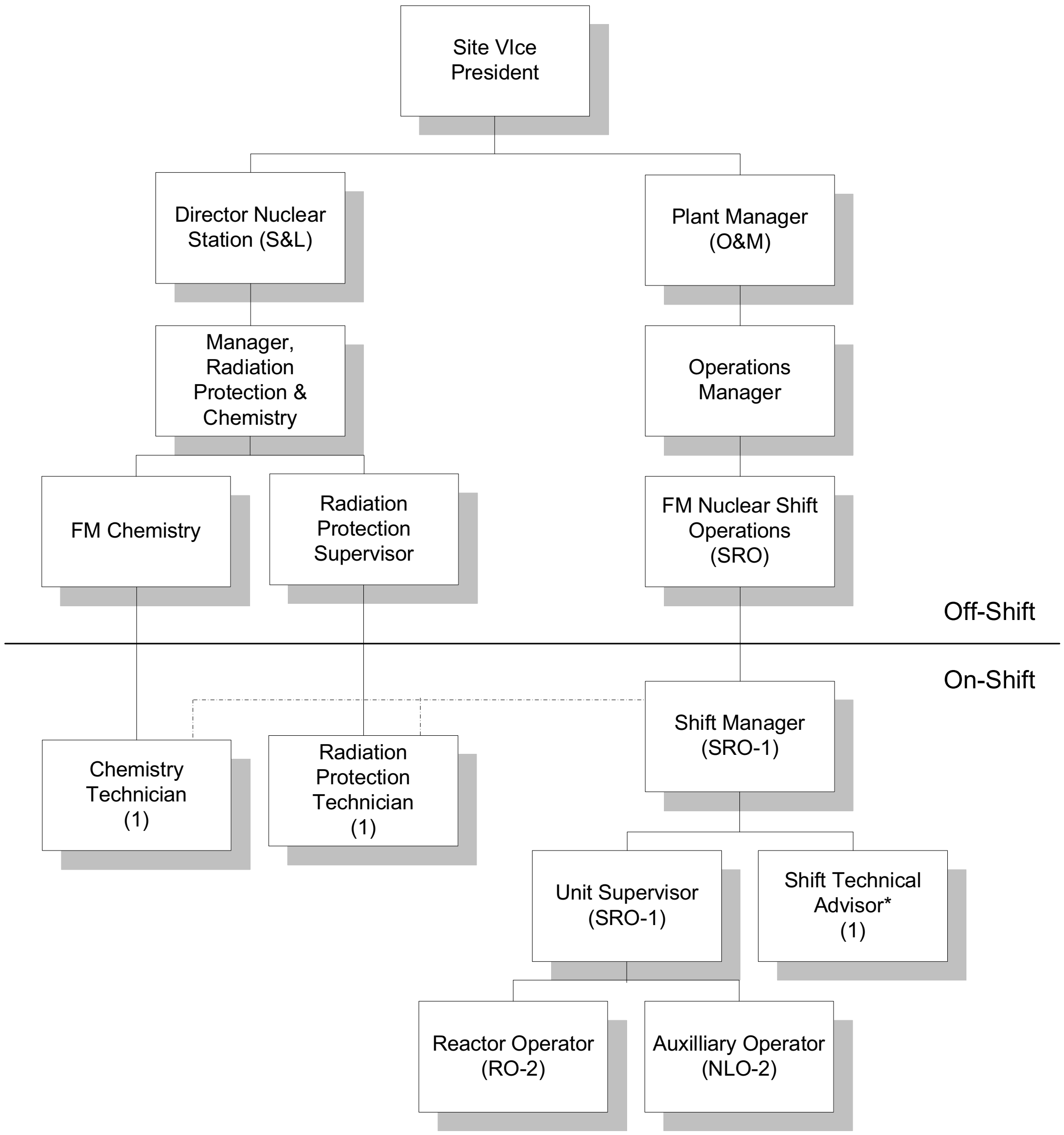


NAPS COL 13.1-1-A

Figure 13.1-202 Nominal Plant Staff Hiring and Training Schedule



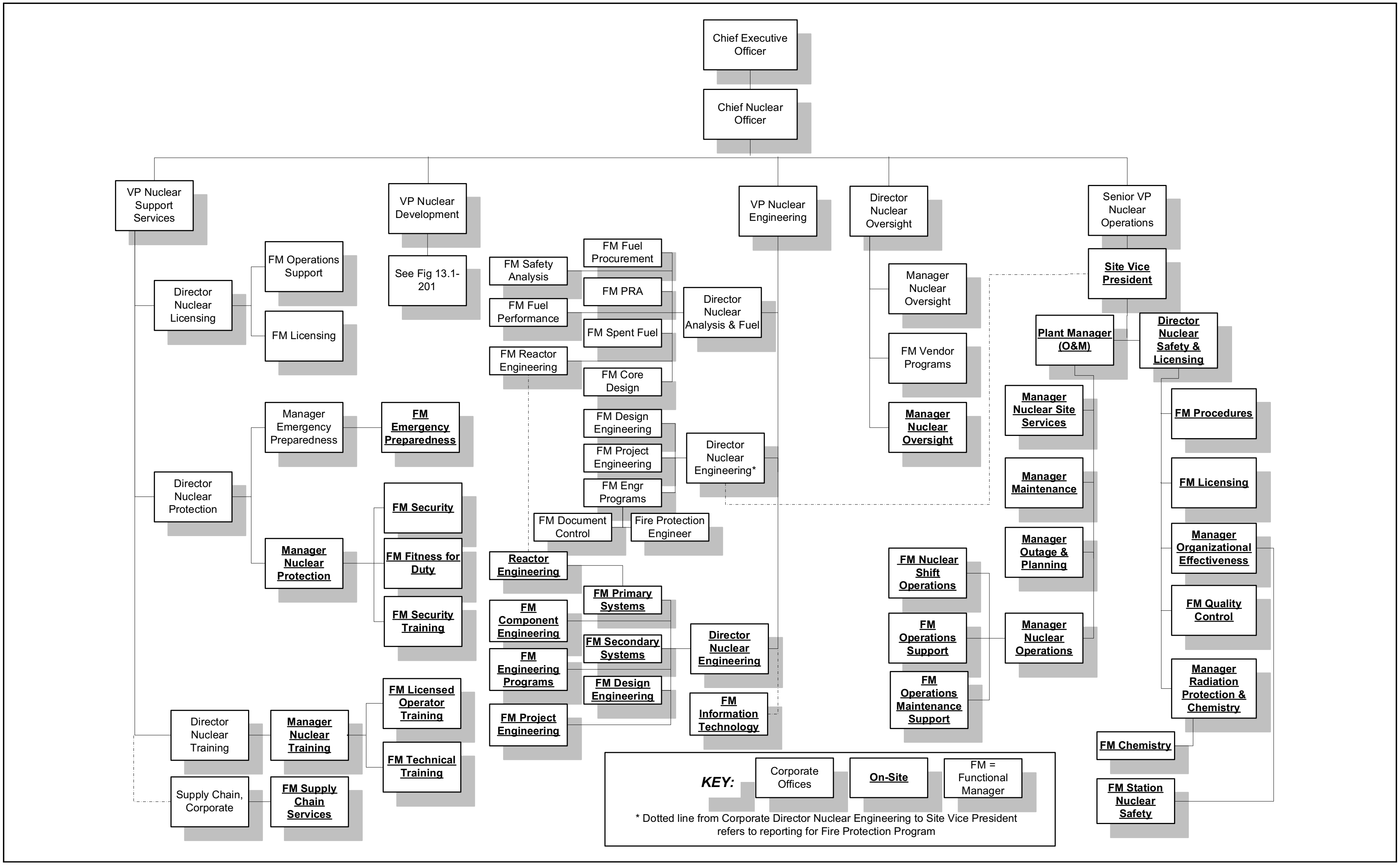
NAPS COL 13.1-1-A **Figure 13.1-203 Shift Operation**



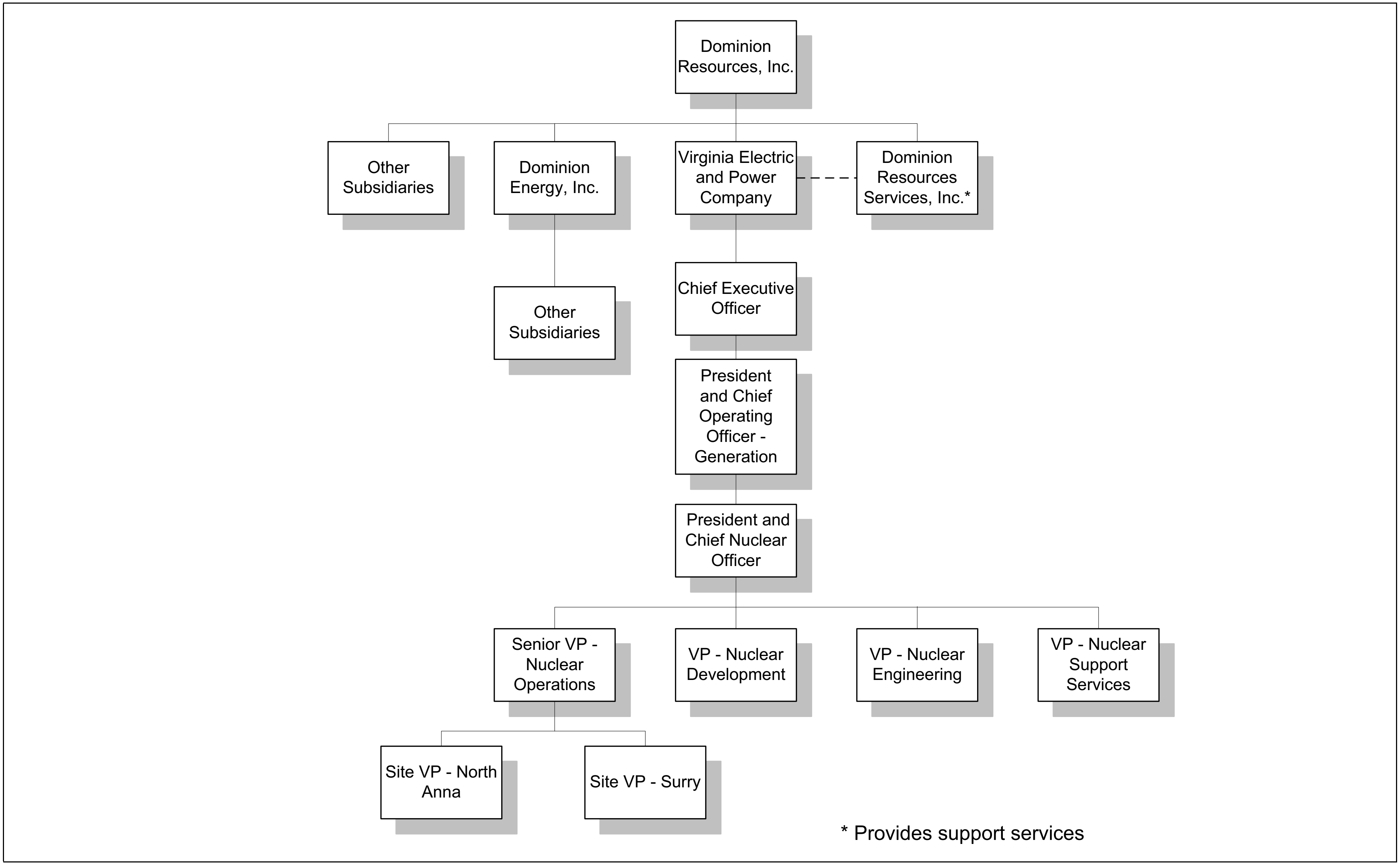
* May be met by onshift combined SRO/STA
SRO - licensed senior reactor operator
RO - licensed reactor operator
NLO - non-licensed operator

Shift Manning - 5 shifts (minimum)
(No.) - Indicates number of positions per shift

NAPS COL 13.1-1-A **Figure 13.1-204 Operating Organization**



NAPS COL 13.1-1-A **Figure 13.1-205 Corporate Structure**



13.2 Training

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

Add the following as introductory material under Section 13.2:

STD SUP 13.2-1 Training programs are addressed in Appendix 13BB. Implementation milestones are addressed in Section 13.4.

13.2.1 Reactor Operator Training

Replace the second sentence of the second paragraph with the following:

STD COL 13.2-1-A Descriptions of the training program and licensed operator requalification program for reactor operators and senior reactor operators are addressed in Appendix 13BB. A schedule showing approximate timing of initial licensed operator training relative to fuel loading is addressed in Section 13.1. Requalification training is implemented in accordance with Section 13.4.

13.2.2 Training for Non-Licensed Plant Staff

Replace the second sentence of the second paragraph with the following:

STD COL 13.2-2-A A description of the training program for non-licensed plant staff is addressed in Appendix 13BB. A schedule showing approximate timing of initial training for non-licensed plant staff relative to fuel load is addressed in Section 13.1.

13.2.5 COL Information

13.2-1-A Reactor Operator Training

STD COL 13.2-1-A This COL item is addressed in Section 13.2.1 and Appendix 13BB.

13.2-2-A Training for Non-Licensed Plant Staff

STD COL 13.2-2-A This COL item is addressed in Section 13.2.2 and Appendix 13BB.

13.3 Emergency Planning

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

Replace the fifth through ninth paragraphs with the following.

STD COL 13.3-1-A

As addressed in the emergency plan, the TSC is provided with reliable voice and data communication with the MCR and Emergency Operations Facility (EOF) and reliable voice communications with the Operational Support Center (OSC), NRC, and state and local operations centers.

The OSC communications system has at least one dedicated telephone extension to the control room, and one dedicated telephone extension to the TSC, and one telephone capable of reaching on-site and off-site locations, as a minimum.

Replace the second sentence in the tenth paragraph with the following.

STD COL 13.3-3-A

Supplies are provided in the service building adjacent to the main change rooms for decontamination of on-site individuals.

13.3.2 Emergency Plan

STD COL 13.3-1-A
STD COL 13.3-2-A
STD COL 13.3-3-A

The emergency plan, prepared in accordance with 10 CFR 52.79(d), is maintained as a separate document.

13.3.3 COL Information

13.3-1-A Identification of OSC and Communication Interfaces with Control Room and TSC

STD COL 13.3-1-A

This COL Item is addressed in Section 13.3 and in Emergency Plan Sections II-F and II-H.

13.3-2-A Identification of EOF and Communication Interfaces with Control Room and TSC

STD COL 13.3-2-A

This COL item is addressed in Section 13.3 and in Emergency Plan Sections II-F and II-H.

13.3-3-A Decontamination Facilities

STD COL 13.3-3-A

This COL item is addressed in Section 13.3 and in Emergency Plan Section II-J.

13.3.5 ESP Information

SSAR Section 13.3 is incorporated by reference for historical purposes.

13.4 Operational Program Implementation

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

Replace this section with the following.

STD COL 13.4-1-A
STD COL 13.4-2-A

Table 13.4-201 lists each operational program, the regulatory source for the program, the associated implementation milestone(s), and the section of the FSAR in which the operational program is fully described as required by RG 1.206, Combined License Applications for Nuclear Power Plants (LWR edition).

13.4.1 COL Information

13.4-1-A Operation Programs

STD COL 13.4-1-A This COL item is addressed in Section 13.4.

13.4-2-A Implementation Milestones

STD COL 13.4-2-A This COL item is addressed in Section 13.4.

13.4.2 References

13.4-201 American Society of Mechanical Engineers (ASME), “Boiler and Pressure Vessel Code (B&PVC), Rules for Inservice Inspection of Nuclear Power Plant Components,” BPVC Section XI.

13.4-202 American Society of Mechanical Engineers (ASME), “Code for the Operation and Maintenance of Nuclear Power Plants,” OM Code.

STD COL 13.4-1-A
STD COL 13.4-2-A

Table 13.4-201 Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	Section	Implementation Milestone	Implementation Requirement
1.	Inservice Inspection Program	10 CFR 50.55a(g) 10 CFR 50.55a(b)(3)(v)	5.2.4 6.6 3.8.1.7.3	Prior to commercial service	10 CFR 50.55a(g) ASME XI IWA 2430(b) (Reference 13.4-201)
	Flow-Accelerated Corrosion Program	10 CFR 50.55a(g)(6)(ii)	6.6.7	Prior to commercial service	License Condition
2.	Inservice Testing Program	10 CFR 50.55a(f)	3.9.6	After generator online on nuclear heat	10 CFR 50.55a(f) ASME OM Code (Reference 13.4-202)
3.	Environmental Qualification Program	10 CFR 50.49(a)	3.11	Prior to fuel load	License Condition
4.	Preservice Inspection Program	10 CFR 50.55a(g)	5.2.4 6.6 3.8.1.7.3	Completion prior to initial plant startup	10 CFR 50.55a(g) ASME Code Section XI IWB/IWC/IWD/IWF-2200(a) (Reference 13.4-201)
5.	Reactor Vessel Material Surveillance Program	10 CFR 50.60 10 CFR 50, Appendix H	5.3.1	Prior to fuel load	License Condition
6.	Preservice Testing Program	10 CFR 50.55a(f)	3.9.6	Prior to fuel load	License Condition
7.	Containment Leakage Rate Testing Program	10 CFR 50.54(o) 10 CFR 50, Appendix J	6.2.6	Prior to fuel load	10 CFR 50, Appendix J Option B – Section III.a

STD COL 13.4-1-A
STD COL 13.4-2-A

Table 13.4-201 Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	Section	Implementation Milestone	Implementation Requirement
8.	Fire Protection Program	10 CFR 50.48	9.5.1.15	Prior to fuel receipt for elements of the Fire Protection Program necessary to support receipt and storage of fuel onsite. Prior to fuel load for elements of the Fire Protection Program necessary to support fuel load and plant operation.	License Condition
9.	Process and Effluent Monitoring and Sampling Program:				
	Radiological Effluent Technical Specifications/Standard Radiological Effluent Controls	10 CFR 20.1301 and 20.1302 10 CFR 50.34a 10 CFR 50.36a 10 CFR 50, Appendix I, Section II and IV	11.5.4.6	Prior to fuel load	License Condition
	Offsite Dose Calculation manual	Same as above	11.5.4.5 11.5.4.8	Prior to fuel load	License Condition
	Radiological Environmental Monitoring Program	Same as above	11.5.4.5	Prior to fuel load	License Condition
	Process Control Program	10 CFR 20.1301 and 20.1302 10 CFR 50.34a 10 CFR 61.55 and 61.56 10 CFR 71	11.4.2.3	Prior to fuel load	License Condition

STD COL 13.4-1-A
STD COL 13.4-2-A

Table 13.4-201 Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	Section	Implementation	
				Milestone	Requirement
10.	Radiation Protection Program	10 CFR 20.1101	12.5	Prior to initial receipt of by-product, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18) for those elements of the Radiation Protection (RP) Program necessary to support such receipt Prior to fuel receipt for those elements of the RP Program necessary to support receipt and storage of fuel onsite Prior to fuel load for those elements of the RP Program necessary to support fuel load and plant operation Prior to first shipment of radioactive waste for those elements of the RP Program necessary to support shipment of radioactive waste	License Condition
11.	Non Licensed Plant Staff Training Program	10 CFR 50.120	13.2.2	18 months prior to scheduled fuel load	10 CFR 50.120(b)

STD COL 13.4-1-A
STD COL 13.4-2-A

Table 13.4-201 Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	Section	Implementation Milestone	Implementation Requirement
12.	Reactor Operator Training Program	10 CFR 55.13 10 CFR 55.31 10 CFR 55.41 10 CFR 55.43 10 CFR 55.45	13.2.1	18 months prior to scheduled fuel load	License Condition
13.	Reactor Operator Requalification Program	10 CFR 50.34(b) 10 CFR 50.54(i) 10 CFR 55.59	13.2	Within 3 months after issuance of an operating license or the date the Commission makes the finding under 10 CFR 52.103(g)	10 CFR 50.54(i-1)
14.	Emergency Planning	10 CFR 50.47 10 CFR 50, Appendix E	13.3	Full participation exercise conducted within 2 years prior to scheduled date for initial loading of fuel	10 CFR Part 50, Appendix E, Section IV.F.2.a(ii)
				Onsite exercise conducted within 1 year prior to the schedule date for initial loading of fuel	10 CFR 50, Appendix E, Section IV.F.2.a(ii)
				Applicant’s detailed implementing procedures for its emergency plan submitted at least 180 days prior to scheduled date for initial loading of fuel	10 CFR 50, Appendix E, Section V

STD COL 13.4-1-A
STD COL 13.4-2-A

Table 13.4-201 Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	Section	Implementation: Milestone	Implementation: Requirement
15.	Security Program:	10 CFR 50.34(c)			
	Physical Security Program	10 CFR 73.55 10 CFR 73.56 10 CFR 73.57	13.6	Prior to fuel receipt	License Condition
	Safeguards Contingency Program	10 CFR 50.34(d) 10 CFR 73, Appendix C	13.6	Prior to fuel receipt	License Condition
	Training and Qualification Program	10 CFR 73, Appendix B	13.6	Prior to fuel receipt	License Condition
	Fitness for Duty (Construction – Mgt & Oversight personnel)	10 CFR 26, Subparts A-H, N, and O	13.7	Prior to on-site construction of safety- or security-related SSCs	License Condition
	Fitness for Duty (Construction – Workers & First Line Supv.)	10 CFR 26 Subpart K	13.7	Prior to on-site construction of safety- or security-related SSCs	License Condition
	Fitness for Duty (Operation)	10 CFR 26	13.7	Prior to fuel receipt	License Condition
16.	Quality Assurance Program – Operation	10 CFR 50.54(a) 10 CFR 50, Appendix A (GDC 1) 10 CFR 50, Appendix B	17.5	30 days prior to scheduled date for initial loading of fuel	10 CFR 50.54(a)(1)
17.	Maintenance Rule	10 CFR 50.65	17.6	Prior to fuel load authorization per 10 CFR 52.103(g)	10 CFR 50.65(a)(1)
18.	Motor-Operated Valve Testing	10 CFR 50.55a(b)(3)(ii)	N/A	There are no safety-related MOVs	

STD COL 13.4-1-A
STD COL 13.4-2-A

Table 13.4-201 Operational Programs Required by NRC Regulations

Item	Program Title	Program Source (Required by)	Section	Implementation Milestone	Implementation Requirement
19.	Initial Test Program	10 CFR 50.34 10 CFR 52.79(a)(28)	14.2	60 days prior to the scheduled date of the first preoperational test for the Preoperational Test Program 60 days prior to the scheduled date of initial fuel loading for the Startup Test Program	License Condition
20.	Snubber Testing and Inspection Program				
	Preservice Inspection Program	10 CFR 50.55a(g) 10 CFR 50.55a(b)(3)(v)	3.9.3.7.1(3)e	Completion prior to initial plant startup	10 CFR 50.55a(g)
	Inservice Inspection Program	10 CFR 50.55a(g) 10 CFR 50.55a(b)(3)(v)	3.9.3.7.1(3)e	Prior to commercial service [a]	10 CFR 50.55a(g) ASME OM Code, ISTD (Reference 13.4-202)
	Inservice Testing Program	10 CFR 50.55a(g) 10 CFR 50.55a(b)(3)(v)	3.9.3.7.1(3)e	After generator online on nuclear heat [a]	10 CFR 50.55a(g) ASME OM Code, ISTD (Reference 13.4-202)
	Preservice Thermal Movement Inspection	10 CFR 50.55a(g) 10 CFR 50.55a(b)(3)(v)	3.9.3.7.1(3)e	During initial heatup and cooldown	10 CFR 50.55a(g) ASME OM Code, ISTD (Reference 13.4-202)
	Preservice Testing Program	10 CFR 50.55a(g) 10 CFR 50.55a(b)(3)(v)	3.9.3.7.1(3)e	Prior to fuel load	License Condition

Notes: a. Snubber inservice examination is initially performed not less than two months after attaining 5% reactor power operation and will be completed within 12 calendar months after attaining 5% reactor power.

13.5 Plant Procedures

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

STD SUP 13.5-1 This section describes the administrative and operating procedures that the operating organization (plant staff) uses to conduct routine operating, abnormal, and emergency activities in a safe manner.

STD SUP 13.5-2 The QAPD describes procedural document control, record retention, adherence, assignment of responsibilities, and changes.

STD SUP 13.5-3 Procedures are identified in this section by topic, type, or classification in lieu of the specific title, and represent general areas of procedural coverage.

STD SUP 13.5-4 Procedures are developed prior to fuel load to allow sufficient time for plant staff familiarization and to allow NRC staff adequate time to review the procedures and to develop operator licensing examinations.

NAPS COL 13.5-4-A Industry guidance for the appropriate format, content, and typical activities delineated in written procedures is implemented, as appropriate. Guidance is based on ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 13.5-202).

STD SUP 13.5-5 The format and content of procedures are controlled by administrative procedure(s). Procedures are organized to include the following components, as necessary:

- Title Page
- Table of Contents
- Scope and Applicability
- Responsibilities
- Prerequisites
- Precautions and Limitations
- Main Body
- Acceptance Criteria
- Check-off Lists
- References

- Attachments and Data Sheets

STD SUP 13.5-6

Each procedure is sufficiently detailed for an individual to perform the required function without direct supervision, but does not provide a complete description of the system or plant process. The level of detail contained in the procedure is commensurate with the qualifications of the individual normally performing the function.

STD SUP 13.5-7

Procedures are developed consistent with guidance described in DCD Section 18.9, Procedure Development, and with input from the human factors engineering process and evaluations.

The bases for procedure development include:

- Plant design bases
- System-based technical requirements and specifications
- Task analyses results
- Risk-important human actions identified in the HRA/PRA
- Initiating events considered in the Emergency Operating Procedures (EOPs), including those events in the design bases
- Generic Technical Guidelines (GTGs) for EOPs

Procedure verification and validation includes the following activities, as appropriate:

- A review to verify they are correct and can be carried out.
- A final validation in a simulation of the integrated system as part of the verification and validation activities as described in DCD Section 18.11, Human Factors Verification and Validation.
- A verification of modified procedures for adequate content, format, and integration. The procedures are assessed through validation if a modification substantially changes personnel tasks that are significant to plant safety. The validation verifies that the procedures correctly reflect the characteristics of the modified plant and can be performed effectively to restore the plant.

STD SUP 13.5-8

Procedures for shutdown management are developed consistent with the guidance described in NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management,” to reduce the potential for loss of reactor coolant system (RCS) boundary and inventory during shutdown conditions. (Reference 13.5-203)

13.5.1 Administrative Procedures

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

STD SUP 13.5-9 This section describes administrative procedures that provide administrative control over activities that are important to safety for the operation of the facility.

Replace the second paragraph with the following:

STD COL 13.5-1-A Administrative procedures are developed in accordance with the nominal schedule presented in Table 13.5-202.

NAPS SUP 13.5-10 Procedures outline the essential elements of the administrative programs and controls as described in ASME NQA-1 and Section 17.5. These procedures are organized such that the program elements are prescribed in documents normally referred to as administrative procedures.

Administrative procedures contain adequate programmatic controls to provide effective interface between organizational elements. This includes contractor and owner organizations providing support to the station operating organization.

NAPS SUP 13.5-11 Procedure control is discussed in the QAPD. Type and content of procedures are discussed throughout Section 13.5.

STD SUP 13.5-12 A procedure style (writer's) guide promotes the standardization and application of human factors engineering principles to procedures. The writer's guide establishes the process for developing procedures that are complete, accurate, consistent, and easy to understand and follow. The guide provides objective criteria so that procedures are consistent in organization, style, and content. The writer's guide includes criteria for procedure content and format including the writing of action steps and the specification of acceptable acronym lists and acceptable terms to be used.

STD SUP 13.5-13 Procedure maintenance and control of procedure updates are performed in accordance with the QAPD.

STD SUP 13.5-14 The administrative programs and associated procedures developed in the pre-COL phase are described in Table 13.5-201 (for future designation as historical information).

STD SUP 13.5-15

13.5.1.1 Administrative Procedures-General

This section describes those procedures that provide administrative controls with respect to procedures, including those that define and provide controls for operational activities of the plant staff.

STD SUP 13.5-16

Plant administrative procedures provide procedural instructions for the following:

- Procedures review and approval
- Procedure adherence
- Scheduling for surveillance tests and calibration
- Log entries
- Record retention
- Containment access
- Bypass of safety function and jumper control
- Communication systems
- Equipment control procedures - These procedures provide for control of equipment, as necessary, to maintain personnel and reactor safety, and to avoid unauthorized operation of equipment
- Control of maintenance and modifications
- Fire Protection Program procedures
- Crane Operation Procedures - Crane operators who operate cranes over fuel pools are qualified and conduct themselves in accordance with ANSI B30.2 (Chapter 2-3), “Overhead and Gantry Cranes” (Reference 13.5-201).
- Temporary changes to procedures
- Temporary procedure issuance and control
- Special orders of a temporary or self-canceling nature
- Standing orders to shift personnel including the authority and responsibility of the shift manager, senior reactor operator in the control room, control room operator, and shift technical advisor
- Manipulation of controls and assignment of shift personnel to duty stations per the requirements of 10 CFR 50.54 (i), (j), (k), (l), and (m) including delineation of the space designated for the “At the Controls” area of the Control Room

- Shift relief and turnover procedures
- Fitness for Duty
- Control Room access
- Working hour limitations
- Feedback of design, construction, and applicable important industry and operating experience
- Shift Manager administrative duties
- Verification of correct performance of operational activities
- A vendor interface program that provides vendor information for safety related components is incorporated into plant documentation

13.5.2 Operating and Maintenance Procedures

Replace the third paragraph with the following:

STD COL 13.5-2-A

Operating Procedures are developed in accordance with Section 13.5.2.1 and Maintenance Procedures are developed in accordance with Section 13.5.2.2.6.1.

Replace the fifth paragraph with the following:

NAPS COL 13.5-4-A

A Plant Operations Procedures Development Plan is established in accordance with Section 13.5.2.1.

Replace the second sentence of "Procedures for Calibration, Inspection and Testing" with the following:

STD COL 13.5-6-H

Surveillance procedures that cover safety-related logic circuitry are addressed in Section 13.5.2.2.6.3.

Replace the second paragraph with the heading "Procedures for Handling of Heavy Loads" with the following:

STD COL 13.5-5-A

The scope of procedures in the Plant Operating Procedures Development Plan is addressed in Section 13.5.2.1.

Replace the last sentence of Section 13.5.2 with the following:

STD COL 13.5-3-A

Emergency Procedures are developed in accordance with Section 13.5.2.1.4.

STD COL 13.5-2-A

13.5.2.1 Operating and Emergency Operating Procedures

This section describes the operating procedures used by the operating organization (plant staff) to conduct routine operating, abnormal, and emergency activities in a safe manner.

Operating procedures are developed at least six months prior to fuel load to allow sufficient time for plant staff familiarization and to allow NRC staff adequate time to review the procedures and to develop operator licensing examinations.

STD SUP 13.5-18

The classifications of operating procedures are:

- System Operating Procedures
- General Operating Procedures
- Abnormal (Off-Normal) Operating Procedures
- Emergency Operating Procedures
- Alarm Response Procedures

STD COL 13.5-2-A

The Plant Operating Procedures Development Plan establishes:

- A scope that includes those operating procedures defined below, which direct operator actions during normal, abnormal, and emergency operations, and considers plant operations during periods when plant systems/equipment are undergoing test, maintenance, or inspection.
- The methods and criteria for the development, verification and validation, implementation, maintenance, and revision of procedures. The methods and criteria are in accordance with NUREG-0737 TMI Items I.C.1 and I.C.9.

STD COL 13.5-5-A

The following procedures are included in the scope of the Plant Operating Procedures Development Plan:

- System operating procedures
- General operating procedures
- Abnormal (off-normal) or alarm response procedures

- Procedures for combating emergencies and other significant events
- Procedures for maintenance and modification
- Procedures for radiation monitoring and control
- Fuel handling procedures
- Temporary procedures
- Procedures for handling of heavy loads

STD COL 13.5-5-A
STD COL 13.5-6-H

- Procedures for calibration, inspection, and testing

NAPS COL 13.5-4-A

Implementation of the Plant Operating Procedures Development Plan establishes:

- Procedures that are consistent with the requirements of 10 CFR 50 and the TMI requirements in NUREG-0737 and Supplement 1 to NUREG-0737
- Requirements that the procedures developed include, as necessary, the elements described in the QAPD
- Bases for specifying plant operating procedures including:
  - Operator actions identified in the vendor's task analysis and PRA efforts in support of the design certification
  - Standardized plant emergency procedure guidelines
  - Consideration of plant-specific equipment selection and site specific elements such as the station water intake structure and the ultimate heat sink
- The definition of the methods through which specific operator skills and training needs, as may be considered necessary for reliable execution of the procedures, are identified and documented
- Requirements that the procedures specified above are made available for the purposes of the Human Factors V&V Implementation Plan described in GE Report NEDO-33276, ESBWR Verification & Validation Implementation Plan (DCD Reference 13.5-1).
- Procedures for the incorporation of the results of operating experience and the feedback of pertinent information into plant procedures in accordance with the provisions of TMI Item I.C.5 (NUREG-0737)

STD SUP 13.5-19

13.5.2.1.1 System Operating Procedures

Instructions for energizing, filling, venting, draining, starting up, shutting down, changing modes of operation, returning to service following testing or maintenance (if not contained in the applicable procedure), and other instructions appropriate for operation of systems are delineated in system procedures.

System procedures contain check-off lists, where appropriate, which are prepared in sufficient detail to provide an adequate verification of the status of the system.

STD SUP 13.5-20

13.5.2.1.2 General Operating Procedures

General operating procedures provide instructions for performing integrated plant operations involving multiple systems such as plant startup and shutdown. These procedures provide a coordinated means of integrating procedures together to change the mode of plant operation or achieve a major plant evolution. Check-off lists are used for the purpose of confirming completion of major steps in proper sequence.

Typical types of general operating procedures are described as follows:

- Startup procedures provide instruction for starting the reactor from cold or hot conditions, establishing power operation, and recovery from reactor trips.
- Shutdown procedures guide operations during and following controlled shutdown or reactor trips, and include instructions for establishing or maintaining hot standby and safe or cold shutdown conditions, as applicable.
- Power operation and load changing procedures provide instruction for steady-state power operation and load changing.

STD SUP 13.5-21

13.5.2.1.3 Abnormal (Off-Normal) Operating Procedures

Abnormal operating procedures for correcting abnormal conditions are developed for those events where system complexity might lead to operator uncertainty. Abnormal operating procedures describe actions to be taken during other than routine operations, which if continued, could lead to either material failure, personnel harm, or other unsafe conditions.

Abnormal procedures are written so that a trained operator knows in advance the expected course of events or indications that identify an abnormal situation and the immediate action to be taken.

NAPS SUP 13.5-22

13.5.2.1.4 Emergency Operating Procedures

EOPs are procedures that direct actions necessary for the operators to mitigate the consequences of transients and accidents that cause plant parameters to exceed reactor protection system or ESF actuation setpoints.

Emergency operating procedures include appropriate guidance for the operation of plant post-72-hour equipment, and are developed as appropriate per the guidance of:

- NUREG-0737, “Clarification of TMI Action Plan Requirements,” Items I.C.1 and I.C.9
- The QAPD

STD COL 13.5-3-A

The emergency operating procedure program (e.g., the procedures generation package (PGP)) describes the objectives of the emergency procedure development process, the program for developing EOPs and the required content of the EOPs.

The procedure development program, as described in the PGP for EOPs, is submitted to the NRC at least three months prior to the planned date to begin formal operator training on the EOPs. The PGP includes:

- GTGs, which are guidelines based on analysis of transients and accidents that are specific to the plant design and operating philosophy. The submitted documentation includes: a) a description of the process used to develop plant-specific technical guidelines (P-STGs) from the GTGs, b) identification of significant deviations from the generic guidelines (including identification of additional equipment beyond that identified in the generic guidelines), along with necessary engineering evaluations or analyses to support the adequacy of each deviation, and c) a description of the process used for identifying operator information and control requirements.
- A plant-specific writer’s guide (P-SWG) that details the specific methods used in preparing EOPs based on P-STGs. The writer’s guide contains objective criteria that require that the emergency procedures developed are consistent in organization, style, content, and usage of terms.
- A description of the program for verification and validation (V&V) of EOPs.
- A description of the program for training operators on EOPs.

- The objectives of the emergency procedure development.
- Discussion of any design change recommendations and/or negative implications that the current design may have on safe operation as noted during implementation of the emergency procedures development plan.

STD SUP 13.5-23

13.5.2.1.5 Alarm Response Procedures

Procedures are provided for annunciators (alarm signals) identifying the proper operator response actions to be taken. Each of these procedures normally contains: a) the meaning of the annunciator or alarm, b) the source of the signal, c) any automatic plant responses, d) any immediate operator action, and e) the long range actions. When corrective actions are very detailed and/or lengthy, the alarm response may refer to another procedure.

NAPS SUP 13.5-24

13.5.2.1.6 Temporary Procedures

Temporary procedures are issued during the operational phase only when permanent procedures do not exist for the following activities: to direct operations during testing, refueling, maintenance, and modifications; to provide guidance in unusual situations not within the scope of the normal procedures; and to provide orderly and uniform operations for short periods when the plant, a system, or a component of a system is performing in a manner not covered by existing detailed procedures, or has been modified or extended in such a manner that portions of existing procedures do not apply.

Temporary operating procedures are developed under established administrative guidelines. They include designation of the period of time during which they may be used and adhere to the QAPD and Technical Specifications, as applicable.

STD SUP 13.5-25

13.5.2.1.7 Fuel Handling Procedures

Fuel handling operations, including fuel receipt, identification, movement, storage, and shipment, are performed in accordance with written procedures. Fuel handling procedures address, for example, the status of plant systems required for refueling; inspection of replacement fuel and control rods; designation of proper tools; proper conditions for spent fuel movement and storage; proper conditions to prevent inadvertent criticality; proper conditions for fuel cask loading and movement; and status of interlocks, reactor trip circuits, and mode switches. These procedures provide instructions for use of refueling equipment, actions for core alterations, monitoring core criticality status, accountability of fuel, and partial or complete refueling operations.

STD SUP 13.5-26

13.5.2.2 Maintenance and Other Operating Procedures

The QAPD provides guidance for procedural adherence.

STD SUP 13.5-27

13.5.2.2.1 Plant Radiation Protection Procedures

The plant radiation protection program is contained in procedures. Procedures are developed and implemented for such things as: maintaining personnel exposures, plant contamination levels, and plant effluents ALARA; monitoring both external and internal exposures of workers, considering industry-accepted techniques; performing routine radiation surveys; performing environmental monitoring in the vicinity of the plant; monitoring radiation levels during maintenance and special work activities; evaluating radiation protection implications of proposed modifications; management of radioactive wastes for offsite shipment, disposal, and treatment; and maintaining radiation exposure records of workers and others.

STD SUP 13.5-28

13.5.2.2.2 Emergency Preparedness Procedures

A discussion of emergency preparedness procedures can be found in the Emergency Plan. A list of implementing procedures is maintained in the Emergency Plan.

STD SUP 13.5-29

13.5.2.2.3 Instrument Calibration and Test Procedures

The QAPD provides a description of procedural requirements for instrumentation calibration and testing.

STD SUP 13.5-30

13.5.2.2.4 Chemistry Procedures

Procedures provided for chemical and radiochemical control activities include the nature and frequency of sampling and analyses; instructions for maintaining fluid quality within prescribed limits; the use of control and diagnostic parameters; and limitations on concentrations of agents that could cause corrosive attack, foul heat transfer surfaces or become sources of radiation hazards due to activation.

Procedures are also provided for the control, treatment, and management of radioactive wastes and control of radioactive calibration sources.

STD SUP 13.5-31

13.5.2.2.5 Radioactive Waste Management Procedures

Procedures for the operation of the radwaste processing systems provide for the control, treatment, and management of on-site radioactive wastes. These procedures are addressed in Section 13.5.2.1.1, System Operating Procedures.

STD SUP 13.5-32
STD COL 13.5-2-A

13.5.2.2.6 Maintenance, Inspection, Surveillance, and Modification Procedures

13.5.2.2.6.1 Maintenance Procedures

Maintenance procedures describe maintenance planning and preparation activities. Maintenance procedures are developed considering the potential impact on the safety of the plant, license limits, availability of equipment required to be operable, and possible safety consequences of concurrent or sequential maintenance, testing, or operating activities.

Maintenance procedures contain sufficient detail to permit the maintenance work to be performed correctly and safely. Procedures include provisions for conducting and recording results of required tests and inspections, if not performed and documented under separate test and inspection procedures. References are made to vendor manuals, plant procedures, drawings, and other sources, as applicable.

Instructions are included, or referenced, for returning the equipment to its normal operating status. Testing is commensurate with the maintenance that has been performed. Testing may be included in the maintenance procedure or be covered in a separate procedure.

Where appropriate sections of related documents, such as vendor manuals, equipment operating and maintenance instructions, or approved drawings with acceptance criteria, provide adequate instructions to provide the required quality of work, the applicable sections of the related documents are referenced in the procedure, or may, in some cases, constitute adequate procedures in themselves. Such documents receive the same level of review and approval as maintenance documents.

The preventive maintenance program, including preventive and predictive procedures, as appropriate, prescribes the frequency and type of maintenance to be performed. An initial program based on service conditions, experience with comparable equipment and vendor recommendations is developed prior to fuel loading. The program is revised and updated as experience is gained with the equipment. To facilitate this, equipment history files are created and maintained. The files are organized to provide complete and easily retrievable equipment history.

STD SUP 13.5-33

13.5.2.2.6.2 Inspection Procedures

The QAPD provides a description of procedural requirements for inspections.

13.5.2.2.6.3 Surveillance Testing Procedures

The QAPD provides a description of procedural requirements for surveillance testing. Surveillance testing procedures are written in a manner that adequately tests all portions of safety-related logic circuitry as described in Generic Letter 96-01, "Testing of Safety Related Logic Circuits."

STD SUP 13.5-34

13.5.2.2.6.4 Modification Procedures

Plant modifications and changes to setpoints are developed in accordance with approved procedures. These procedures control necessary activities associated with the modifications such that they are carried out in a planned, controlled, and orderly manner. For each modification, design documents such as drawings, equipment and material specifications, and appropriate design analyses are developed, or the as-built design documents are utilized. Separate reviews are conducted by individuals knowledgeable in both technical and QA requirements to verify the adequacy of the design effort.

Proposed modifications that involve a license amendment or a change to Technical Specifications are processed as proposed license amendment request.

Plant procedures impacted by modifications are changed to reflect revised plant conditions prior to declaring the system operable and cognizant personnel who are responsible for operating and maintaining the modified equipment are adequately trained.

STD SUP 13.5-35

13.5.2.2.6.5 Heavy Load Handling Procedures

This topic is discussed in Section 9.1.5.8.

STD SUP 13.5-36

13.5.2.2.7 Material Control Procedures

The QAPD provides a description of procedural requirements for material control.

STD SUP 13.5-37

13.5.2.2.8 Security Procedures

A discussion of security procedures is provided in the Security Plan.

STD SUP 13.5-38

13.5.2.2.9 Refueling and Outage Planning Procedures

Procedures provide guidance for the development of refueling and outage plans, and as a minimum address the following elements:

- An outage philosophy which includes safety as a primary consideration in outage planning and implementation
- Separate organizations responsible for scheduling and overseeing the outage and provisions for an independent safety review team that would be assigned to perform final review and grant approval for outage activities
- Control procedures, which address both the initial outage plan and safety-significant changes to schedule
- Provisions that activities receive adequate resources
- Provisions that defense-in-depth during shutdown and margins are not reduced or provisions that an alternate or backup system must be available if a safety system or a defense-in-depth system is removed from service
- Provisions that personnel involved in outage activities are adequately trained including operator simulator training to the extent practicable, and training of other plant personnel, including temporary personnel, commensurate with the outage tasks they are to perform
- The guidance described in NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management,” to reduce the potential for loss of reactor coolant system boundary and inventory during shutdown conditions (Reference 13.5-203)

13.5.3 COL Information

13.5-1-A Administrative Procedures Development Plan

STD COL 13.5-1-A

This COL item is addressed in Section 13.5.1.

13.5-2-A Plant Operating Procedures Development Plan

STD COL 13.5-2-A This COL item is addressed in Section 13.5.2.

13.5-3-A Emergency Procedures Development

STD COL 13.5-3-A This COL item is addressed in Section 13.5.2.

13.5-4-A Implementation of the Plant Procedures Plan

NAPS COL 13.5-4-A This COL item is addressed in Section 13.5 and Section 13.5.2.

13.5-5-A Procedures Included in Scope of Plan

STD COL 13.5-5-A This COL item is addressed in Section 13.5.2.

13.5-6-H Procedures for Calibration, Inspection, and Testing

STD COL 13.5-6-H This COL item is addressed in Section 13.5.2.

13.5.4 References

13.5-201 American National Standards Institute, Overhead and Gantry Cranes, ANSI B30.2- 2001.

13.5-202 American Society of Mechanical Engineers, Quality Assurance Requirements for Nuclear Facility Applications, NQA-1-1994.

13.5-203 Nuclear Utilities Management and Resources Council, Guidelines for Industry Actions to Assess Shutdown Management, NUMARC 91-06, December 1991.

13.5-204 General Electric Corporation, Licensing Topical Report ESBWR Human Factors Engineering Procedures Development Implementation Plan, NEDO-33274, Revision 2, March 2007.

STD SUP 13.5-39

Table 13.5-201 Pre-COL Phase Administrative Programs and Procedures

(This table is included for future designation as historical information.)

Design/Construction Quality Assurance Program
Reporting of Defects and Noncompliance,10 CFR 21 Program
Construction License Fitness for Duty Programs, 10 CFR 26
Design Reliability Assurance Program

STD COL 13.5-1-A

Table 13.5-202 Nominal Procedure Development Schedule

(This table is included for future designation as historical information.)

Category A: Controls

Group	Procedure Type	Preparation Milestone
1	Procedures review and approval	6 months before first license class
2	Equipment control procedures	18 months before fuel load
3	Control of maintenance and modifications	18 months before fuel load
4	Fire Protection procedures	1. 6 months before fuel receipt for elements of the program supporting fuel onsite 2. 6 months before fuel load for elements supporting fuel load and plant operation
5	Crane operation procedures	6 months before fuel receipt
6	Temporary changes to procedures	6 months before first license class
7	Temporary procedures	6 months before first license class
8	Special orders of a transient or self-canceling character	6 months before first license class

Category B: Specific Procedures

Group	Procedure Type	Preparation Milestone
1	Standing orders to shift personnel including the authority and responsibility of the shift supervisor, licensed senior reactor operator in the control room, control room operator, and shift technical advisor	6 months before first license class
2	Assignment of shift personnel to duty stations and definition of “surveillance area”	6 months before first license class
3	Shift relief and turnover	6 months before fuel load
4	Fitness for duty	1. Construction FFD program: 6 months before on-site construction of safety- or security-related SSCs 2. Operational FFD program: 6 months before fuel load
5	Control room access	6 months before fuel load

STD COL 13.5-1-A

Table 13.5-202 Nominal Procedure Development Schedule

6	Limitations on work hours	6 months before fuel load
7	Feedback of design, construction, and applicable important industry and operating experience	6 months before fuel load
8	Shift supervisor administrative duties	6 months before fuel load
9	Verification of correct performance of operating activities	6 months before first license class

13.6 Physical Security

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

13.6.1.1.3 Detection Aids

Replace the last sentence in the third paragraph with the following.

STD COL 13.6-9-A

Operating alarm response procedures will be developed and implemented in accordance with milestone defined in Section 13.5.2.1.

Replace the last sentence in the fourth paragraph with the following.

STD COL 13.6-13-A

This action will be completed prior to the milestone for Physical Security Plan implementation (Table 13.4-201).

13.6.1.1.5 Access Controls

Replace the first sentence in the third paragraph with the following.

STD COL 13.6-6A

A key control program will be developed and implemented prior to the milestone for Physical Security Plan implementation (Table 13.4-201).

Replace the fifth paragraph with the following.

STD COL 13.6-14-A

Administrative procedures will be developed prior to the milestone for Physical Security Plan implementation (Table 13.4-201) to control work being performed in cabinets containing the control circuitry (contact elements) for the systems listed in Table 4-1 of NEDE-33391.

Replace the last sentence in the sixth paragraph with the following.

STD COL 13.6-15-A

Administrative procedures will be developed prior to the milestone for Physical Security Plan implementation (Table 13.4-201) that will require two persons, each of whom are qualified to perform the intended work, to be present during the performance of any work on systems listed in Table 4-1 of NEDE-33391.

13.6.1.1.8 Testing

Replace the last sentence in the first paragraph with the following.

STD COL 13.6-10-A The establishment of these surveillance test procedures and frequencies will be completed in accordance with the milestone for Physical Security Plan implementation (Table 13.4-201).

Replace the last sentence in the second paragraph with the following.

STD COL 13.6-11-A The establishment of these testing and maintenance milestones will be completed in accordance with the milestone for Physical Security Plan implementation (Table 13.4-201).

STD COL 13.6-8-H The licensee will demonstrate through a one time test, analysis, or a combination of tests and analyses, that no single postulated security event will disable the capability of both the Central and Secondary Alarm Stations. This demonstration will be completed prior to the milestone for Physical Security Plan implementation (Table 13.4-201).

13.6.2 Security Plan

Add the following at the end of this section:

STD SUP 13.6-1 The Physical Security Plan during construction, including control of access to the new plant construction site, is consistent with NEI 03-12, Appendix F (Reference 13.6-201), which is currently under NRC review. Table 13.4-201 provides milestones for security program implementation.

NAPS ESP COL 13.6-1 The design requirements for protected area barriers are described in the Physical Security Plan. The barriers will be designed and located to support the security response strategy timelines. The specific designs for protected area barriers will be completed as part of detailed plant design before the milestone for Physical Security Plan implementation (Table 13.4-201).

STD COL 13.6-12-A As part of the Security Plan, the licensee will develop an integrated response strategy to a confirmed security event that provides for manual actuation of plant systems by the operators to an evolving scenario necessitating escalating operator response. This action will be completed prior to the milestone for Physical Security Plan implementation (Table 13.4-201).

13.6.3 COL Information

13.6-6-A Key Control

STD COL 13.6-6-A This COL item is addressed in Section 13.6.1.1.5.

13.6-7-A Secondary Alarm Station Design

STD COL 13.6-7-A This COL item is addressed in the Physical Security Plan.

13.6-8-H CAS and SAS Redundancy

STD COL 13.6-8-H This COL item is addressed in Section 13.6.1.1.8.

13.6-9-A Operational Alarm Response Procedures

STD COL 13.6-9-A This COL item is addressed in Section 13.6.1.1.3.

13.6-10-A Operational Surveillance Test Procedures

STD COL 13.6-10-A This COL item is addressed in Section 13.6.1.1.8.

13.6-11-A Maintenance Test Procedures

STD COL 13.6-11-A This COL item is addressed in Section 13.6.1.1.8.

13.6-12-A Operational Response Procedures to Security Events

STD COL 13.6-12-A This COL item is addressed in Section 13.6.2.

13.6-13-A Operational Alarm Response Procedures

STD COL 13.6-13-A This COL item is addressed in Section 13.6.1.1.3.

13.6-14-A Administrative Controls to Sensitive Cabinets

STD COL 13.6-14-A This COL item is addressed in Section 13.6.1.1.5.

13.6-15-A Administrative Controls to Sensitive Equipment

STD COL 13.6-15-A This COL item is addressed in Section 13.6.1.1.5.

13.6.4 References

13.6-201 Nuclear Energy Institute, Security Measures During New Reactor Construction, NEI 03-12 Appendix F.

NAPS SUP 13.6-2

13.6.5 ESP Information

SSAR Section 13.6 is incorporated by reference.

STD SUP 13.7-1

13.7 Fitness For Duty

The Fitness for Duty (FFD) Program is implemented and maintained in two phases: the construction phase program and the operating phase program. The construction phase program is consistent with NEI 06-06 (Reference 13.7-201), which is currently under NRC review. The construction phase program is implemented, as identified in Table 13.4-201, prior to on-site construction of safety- or security-related SSCs. The operations phase program is consistent with NEI 03-01 (Reference 13.7-202), which is currently under NRC review. The operations phase program is implemented prior to fuel receipt, as identified in Table 13.4-201.

13.7.1 References

13.7-201 Nuclear Energy Institute (NEI) “Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites,” NEI 06-06.

13.7-202 Nuclear Energy Institute (NEI) “Nuclear Power Plant Access Authorization Program,” NEI 03-01.

NAPS COL 13.1-1-A

Appendix 13AA Design and Construction Responsibilities

13AA.1 Design and Construction Activities

Dominion has substantial experience in the design, construction, and operation of nuclear power plants and substantial experience in activities of similar scope and complexity. Dominion was responsible for the design and construction activities associated with two existing nuclear power stations in Virginia, Surry and North Anna, both of which Dominion currently operates. Dominion oversaw the activities of Westinghouse Electric Company and Stone & Webster Engineering Corporation in the design and construction of those stations.

In addition, Dominion has been responsible for the design, construction, and operation of several large fossil stations, activities of similar scope and complexity. One example is Chesterfield Power Station in Virginia. Dominion oversaw the activities of Combustion Engineer, General Electric Co. and Stone & Webster in the design and construction of the station. Dominion currently operates Chesterfield Power Station. The station generates over 1700 MWe.

Dominion's management, engineering, and technical support organization for the construction and operation of Unit 3 are described in Chapters 17 and 13, respectively. As described in Section 1.4.1, Dominion has selected GEH as its primary contractor for the design of Unit 3, and Bechtel as the primary contractor for site engineering. The contractors for the construction of the nuclear island and the turbine island have not yet been selected.

Other design and construction activities will be contracted to qualified suppliers of such services. Implementation or delegation of design and construction responsibilities is described in the sections below. Quality Assurance aspects are described in Chapter 17.

13AA.1.1 Principal Site-Related Engineering Work

The principal site engineering activities accomplished towards the construction and operation of the plant are:

Meteorology

Information concerning local (site) meteorological parameters is developed and applied by station and contract personnel to assess the impact of the station on local meteorological conditions. An onsite meteorological measurements program is employed by station personnel to produce data for the purpose of making atmospheric dispersion estimates for postulated accidental and expected routine airborne releases of effluents. A maintenance program is established for surveillance, calibration, and repair of instruments. More information regarding the study and meteorological program is found in Section 2.3.

Geology

Information relating to site and regional geotechnical conditions is developed and evaluated by utility and contract personnel to determine if geologic conditions could present a challenge to safety of the plant. Items of interest include geologic structure, seismicity, geological history, and ground water conditions. The excavation for safety-related structures will be geologically mapped and photographed by experienced geologists. Unforeseen geologic features that are encountered will be evaluated. Section 2.5 provides details of these investigations.

Seismology

Information relating to seismological conditions is developed and evaluated by utility and contract personnel to determine if the site location and area surrounding the site is appropriate from a safety standpoint for the construction and operation of a nuclear power plant. Information regarding tectonics, seismicity, correlation of seismicity with tectonic structure, characterization of seismic sources, and ground motion are assessed to estimate the potential for strong earthquake ground motions or surface deformation at the site. Section 2.5 provides details of these investigations.

Hydrology

Information relating to hydrological conditions at the plant site and the surrounding area is developed and evaluated by utility and contract personnel. The study includes hydrologic characteristics of streams, lakes, shore regions, the regional and local groundwater environments, and existing or proposed water control structures that could influence flood control and plant safety. Section 2.4 includes more detailed information regarding this subject.

Demography

Information relating to local and surrounding area population distribution is developed and evaluated by utility and contract personnel. The data is used to determine if requirements are met for establishment of exclusion area, low population zone, and population center distance. Section 2.1 includes more detailed information regarding population around the plant site.

Environmental Effects

Monitoring programs are developed to enable the collection of data necessary to determine possible impact on the environment due to construction, startup, and operational activities and to establish a baseline from which to evaluate future environmental monitoring. This program is described in the ESP-ER and in COLA Part 3.

13AA.1.2 **Design of Plant and Ancillary Systems**

Design and construction of systems outside the power block such as circulating water, service water, switchyard, and secondary fire protection systems are performed by Dominion or qualified contractors, as assigned.

13AA.1.3 **Review and Approval of Plant Design Features**

Design engineering review and approval is performed in accordance with Chapter 17. The reactor vendor is responsible for design control of the power block. Design work is performed in accordance with the design and construction QA manual including the reviews necessary to verify the adequacy of the design. Verification is performed by competent individuals or groups other than those who performed the original design. Design issues arising during construction are addressed and implemented with notification and communication of changes to the manager in charge of engineering for review. As systems are tested and approved for turnover and operation, control of design is turned over to plant staff. The manager in charge of engineering, along with functional managers and staff, assumes responsibility for review and approval of modifications, additions, or deletions in plant design features, as well as control of design documentation, in accordance with the Operational QA Program. Design control becomes the responsibility of the manager in charge of engineering prior to loading fuel. During construction, startup, and operation, changes to human-system interfaces of control room design are approved using a Human Factors Engineering evaluation addressed within DCD Chapter 18. See Figure 13.1-201, Construction Organization and the QAPD (incorporated into Section 17.5) for reporting relationships.

13AA.1.4 Environmental Effects

Impact to the surrounding environment from construction and operating activities is fully addressed in COLA Part 3, Applicants' Environmental Report - Combined License Stage.

13AA.1.5 Security Provisions

The Physical Security Plan is designed with provisions that meet the applicable NRC regulations. See Section 13.6 and the Security Plan, which was submitted under separate transmittal.

13AA.1.6 Development of Safety Analysis Reports

Information regarding the development of the FSAR is found in Chapter 1.

13AA.1.7 Review and Approval of Material and Component Specifications

Safety-related material and component specifications of SSCs designed by the reactor vendor are reviewed and approved in accordance with the reactor vendor quality assurance program and Section 17.1. Review and approval of items not designed by the reactor vendor are controlled for review and approval by Section 17.5 and the QAPD.

13AA.1.8 Procurement of Materials and Equipment

Procurement of materials during construction phase is the responsibility of the reactor vendor and constructor. The process is controlled by the construction QA programs of these organizations. Oversight of the inspection and receipt of materials process is the responsibility of the manager in charge of nuclear oversight.

13AA.1.9 Management and Review of Construction Activities

Management and responsibility for construction activities is assigned to the construction manager. This position reports to the Engineer, Procure, and Construct (EPC) executive, who is accountable to the CNO. See Figure 13.1-201, Construction Organization.

Monitoring and review of construction activities by utility personnel is a continuous process at the plant site. Contractor performance is monitored to provide objective data to utility management in order to identify problems early and develop solutions. Monitoring of construction activities verifies that the contractors are in compliance with contractual obligations for quality, schedule, and cost. To maintain independence from the construction organization, the oversight organization reports directly to the CNO.

Monitoring and review of construction activities is divided functionally across the various disciplines of the utility construction staff, i.e. electrical, mechanical, instrument and control, etc., and tracked by schedule based on system and major plant components/areas.

After each system is turned over to plant staff the construction organization relinquishes responsibility for that system. At that time the construction organization will be responsible for completion of construction activities as directed by plant staff and available to provide support for start-up testing as necessary.

13AA.2 Preoperational Activities

This section describes the activities required to transition the unit from the construction phase to the operational phase. These activities include turnover of systems from construction, preoperational testing, schedule management, test procedure development, fuel load, integrated startup testing, and turnover of systems to plant staff.

13AA.2.1 Development of Human Factors Engineering Design Objectives and Design Phase Review of Proposed Control Room Layouts

HFE design objectives are initially developed by the reactor vendor in accordance with DCD Chapter 18. As a collaborative team, personnel from the reactor vendor design staff and personnel, including licensed operators, engineers, and instrumentation and control technicians from owner and other organizations in the nuclear industry, assess the design of the control room and man-machine interfaces to attain safe and efficient operation of the plant. See DCD Section 18.2 for additional details of HFE program management.

Modifications to the certified design of the control room or man-machine interface described in the DCD are reviewed per engineering procedures, as required by DCD Section 18.2, to evaluate the impact to plant safety. The engineering manager is responsible for the human factors engineering design process and for the design commitment to HFE during construction and throughout the life of the plant. The HFE program is established in accordance with the description and commitments in DCD Chapter 18.

13AA.2.2 **Preoperational and Startup Testing**

Functional managers reporting to the plant manager are assigned responsibility for organizing and developing the preoperational testing and startup testing organizations. These organizations prepare procedures and schedules and conduct preoperational and startup testing. The preoperational and startup testing organizations are staffed by testing engineers, procedure writers, and planner/schedulers. The qualification requirements of testing engineers in the preoperational and startup testing organizations meet those established in ANSI/ANS-3.1 (Reference 13.1-201).

Test engineers are responsible for integrated testing of systems to prove functionality of system design requirements. They provide guidance and supervision to procedure writers and communicate closely with operations personnel and other supporting staff to facilitate safe and efficient performance of preoperational and startup tests. The scope of testing to be accomplished is presented in Chapter 14. As systems are turned over from the constructor they are tested by component then by integrated system preoperational test. Sufficient numbers of personnel are assigned to perform preoperational and startup testing to facilitate safe and efficient implementation of the testing program. Plant-specific training provides instruction on the administrative controls of the test program. The startup test program provides data and experience useful during the operational phase.

During the preoperational and startup testing phases, the constructor and reactor vendor staff support, as necessary, the testing performed by the nuclear plant preoperational and startup testing staffs. The functional managers in charge of preoperational and startup testing are assisted by other station organizations including operations, plant maintenance, and engineering. These assisting organizations provide support in developing test procedures, conducting the test program, and in reviewing test results.

Procedures are written to describe organizational responsibilities and interfaces between staff, constructor, and reactor vendor, and to establish direction in writing, reviewing, and performing tests. The construction organization, depicted in Figure 13.1-201, includes the preoperational and startup testing functional groups.

13AA.2.3 Development and Implementation of Staff Recruiting and Training Programs

Staffing plans are developed with input from the reactor vendor for safe operation of the plant as determined by HFE. See DCD Section 18.6. These plans are developed under the direction and guidance of the Vice President - Nuclear Development (see Table 13.1-201 and Figure 13.1-201). Staffing plans will be completed and manager level positions filled prior to start of preoperational testing. Personnel selected to be licensed reactor operators and senior reactor operators along with other staff necessary to support the safe operation of the plant are hired with sufficient time available to complete appropriate training programs and become qualified and licensed (if required) prior to fuel being loaded in the reactor vessel. See Figure 13.1-202 for hiring and training requirements for operator and technical staff relative to fuel load.

Table 13.1-201 includes the initial estimated number of staff for selected positions that will be filled at the time of initial fuel load. Recruiting of personnel to fill positions is the shared responsibility of the manager in charge of human resources and the various heads of departments. The training program is described in Section 13.2.

13AA.2.4 Transition to Operating Phase

The construction executive (Vice President - Nuclear Development) is responsible for developing and implementing a plan for the organizational transition from the construction phase to the operating phase. The plan is fully implemented and transition completed prior to commencement of commercial operations with operational responsibility then fully under the direction of the Senior Vice President - Nuclear Operations.

Appendix 13BB Training Program

STD SUP 13.2-1
STD COL 13.2-1-A
STD COL 13.2-2-A

NEI 06-13A (Reference 13BB-201), Technical Report on a Template for an Industry Training Program Description, which is under review by the NRC staff, is incorporated by reference.

13BB References

13BB-201 Nuclear Energy Institute (NEI), “Technical Report on a Template for an Industry Training Program Description,” NEI 06-13A.

Chapter 14 Initial Test Program

14.1 Initial Test Program for Preliminary Safety Analysis Reports

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

14.2 Initial Plant Test Program for Final Safety Analysis Reports

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

14.2.1.4 Organization and Staffing

Add the following at the end of this section.

NAPS SUP 14.2-1

Section 13.1 provides additional information regarding responsibilities, qualifications, and organization for implementing the pre-operational and startup testing program.

14.2.2.1 Startup Administrative Manual

Replace the first two paragraphs with the following.

STD COL 14.2-1-A
STD COL 14.2-2-H

A description of the Initial Test Program (ITP) administration is provided in Appendix 14AA. The Startup Administrative Manual (SAM) will be developed and made available for review 60 days prior to scheduled start of the preoperational test program.

14.2.2.2 Test Procedures

Replace the last two sentences in this section with the following.

STD COL 14.2-3-H

Approved test procedures for satisfying the commitments of this section will be developed and available for review no later than 60 days prior to their intended use for preoperational tests and no later than 60 days prior to scheduled fuel loading for power ascension tests.

14.2.2.5 Test Records

Add the following at the end of this section.

STD SUP 14.2-2

Startup test reports are prepared in accordance with RG 1.16.

14.2.7 Test Program Schedule and Sequence

Replace the last paragraph with the following.

STD COL 14.2-4-H

The detailed testing schedule will be developed and made available for review prior to actual implementation. The schedule may be updated and continually optimized to reflect actual progress and subsequent revised projections.

The implementation milestones for the Initial Test Program are provided in Section 13.4.

14.2.8.1.36 AC Power Distribution System Preoperational Test General Test Methods and Acceptance Criteria

Add the following at the end of this section.

STD-SUP-14.2-4

- Proper operation of the automatic transfer capability of the normal preferred power source to the alternate preferred power source.

14.2.9 Site-Specific Preoperational and Startup Tests

Replace the second and third paragraphs with the following.

NAPS COL 14.2-5-A

This section describes the site specific pre-operational and initial startup tests not addressed in DCD Section 14.2.8.

NAPS COL 14.2-6-H

Specific testing to be performed and the applicable acceptance criteria for each preoperational and startup test are documented in test procedures to be made available to the NRC approximately 60 days prior to their intended use for preoperational tests, and not less than 60 days prior to scheduled fuel load for initial startup tests. Site-specific preoperational tests are in accordance with the system specifications and associated equipment specifications for equipment in those systems provided by the licensee that are not part of the standard plant described in DCD Section 14.2.8. The tests demonstrate that the installed equipment and systems perform within the limits of these specifications.

14.2.9.1 Site-specific Pre-Operational Tests

Replace this section with the following.

NAPS SUP 14.2-3

14.2.9.1.1 Station Water System Pre-Operation Test

Purpose

The objective of this test is to verify proper operation of the SWS and its ability to supply design quantities and quality of water to the CIRC, PSWS cooling tower basin, MWS, and FPS.

Prerequisites

The construction tests have been successfully completed and the SCG has reviewed the test procedure and approved the initiation of testing. Electrical power, the CIRC, PSWS, MWS and FPS, instrument air, Chemical Storage and Transfer System, and other required interfacing systems are available, as needed, to support the specified testing.

General Test Methods and Acceptance Criteria

Performance is observed and recorded during a series of individual component and integrated system tests to demonstrate the following:

- Proper operation of instrumentation and equipment in appropriate design combinations of logic and instrument channel trip;
- Proper functioning of instrumentation and alarms used to monitor system operation and availability;
- Proper operation of pumps, motors, and valves in all design operating modes;
- Proper operation of motorized self-cleaning strainers;
- Proper system flow paths and flow rates, including pump capacity and discharge head;
- Proper operation of interlocks and equipment protective device in pump, motor, and valve controls;
- Proper operation of freeze protection methods and devices, where installed; and
- Acceptability of pump/motor vibration levels.

14.2.9.1.2 Cooling Tower Preoperational Test

Purpose

The objective of this test is to verify proper operation of the waste heat rejection portion of the CIRC (i.e., the dry cooling array and the hybrid cooling tower and basin.) Testing of the balance of the CIRC is addressed in DCD Section 14.2.8.1.50.

Prerequisites

The construction tests have been successfully completed and the SCG has reviewed the test procedure and approved the initiation of testing. Electrical power, the CIRC, SWS, Instrument Air System, Chemical Storage and Transfer System, and other required interfacing systems are available, as needed, to support the specific testing.

General Test Methods and Acceptance Criteria

Because of insufficient heat loads during the preoperational test phase, cooling tower performance evaluations are performed during the startup phase with the turbine generator on line.

Operation is observed and recorded during a series of individual component and integrated system tests to demonstrate the following:

- Proper operation of instrumentation and equipment in appropriate design combinations of logic and instrument channel trip;
- Proper functioning of instrumentation and alarms used to monitor system operation and availability;
- Proper operation of pumps, fans, motors, and valves in all design operating modes;
- Proper system flow paths and flow rates, including pump capacity and discharge head;
- Proper operation of interlocks and equipment protective devices in pump, motor, and valve controls;
- Proper operation of freeze protection methods and devices, where installed; and
- Acceptability of pump/motor vibration levels.

14.2.9.1.3 **[Deleted]**

14.2.9.1.4 **[Deleted]**

14.2.9.2 **Site-Specific Startup Tests**

Replace this section with the following.

NAPS SUP 14.2-2

14.2.9.2.1 **Cooling Tower Performance Test**

Purpose

The objective of this test is to demonstrate acceptable performance of the waste heat rejection portion of the CIRC (i.e., the dry cooling array and the hybrid cooling tower and basin), particularly its ability to cool design quantities of circulating water to design temperature under expected operational load conditions.

Prerequisites

The preoperational tests are complete and plant management has reviewed the test procedure and approved the initiation of testing. The plant is in the appropriate operational configuration for the scheduled testing. The necessary instrumentation is checked or calibrated.

Description

Power ascension phase testing of the waste heat rejection portions of the CIRC is necessary to the extent that fully loaded conditions could not be approached during the preoperational phase. Pertinent parameters are monitored in order to provide a verification of proper system flow balancing and performance of both the dry cooling array and hybrid-cooling tower.

Criteria

System performance is consistent with design requirements.

14.2.10 **COL Information**

14.2-1-A **Description - Initial Test Program Administration**

STD COL 14.2-1-A This COL Item is addressed in Section 14.2.2.1 and Appendix 14AA.

14.2-2-H **Startup Administrative Manual**

STD COL 14.2-2-H This COL Item is addressed in Section 14.2.2.1.

14.2-3-H **Test Procedures**

STD COL 14.2-3-H This COL Item is addressed in Section 14.2.2.2.

14.2-4-H Test Program Schedule and Sequence

NAPS COL 14.2-4-H

This COL Item is addressed in Section 14.2.7.

14.2-5-A Site Specific Tests

NAPS COL 14.2-5-A

This COL Item is addressed in Section 14.2.9.

14.2-6-H Site Specific Test Procedures

NAPS COL 14.2-6-H

This COL Item is addressed in Section 14.2.9.

14.3 Inspections, Tests, Analysis, and Acceptance Criteria

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

14.3.8 Overall ITAAC Content for Combined License Applications

Replace the last paragraph with the following.

STD COL 14.3-1-A

The requirements for inclusion of Emergency Planning ITAAC (EP-ITAAC) in a COLA are provided in 10 CFR 52.80(a). In SRM-SECY-05-0197, the NRC-approved generic EP-ITAAC for use in COL and ESP applications. This set of EP-ITAAC was considered in the development of the plant-specific EP-ITAAC, which are tailored to the ESBWR design. The plant-specific EP-ITAAC are included in a separate part of the COLA.

14.3.9 Site-Specific ITAAC

Delete the last sentence of the first paragraph and add the following at the end of this section.

STD COL 14.3-2-A

The selection criteria and methodology provided in this section of the referenced DCD were utilized as the site-specific selection criteria and methodology for ITAAC. These criteria and methodology were applied to those site-specific (SS) systems that were not evaluated in the referenced DCD. The entire set of ITAAC for the facility, including DC-ITAAC, EP-ITAAC, PS-ITAAC, and SS-ITAAC, is included in a separate part of the COLA.

14.3.10 COL Information

14.3-1-A Emergency Planning ITAAC

STD COL 14.3-1-A

This COL item is addressed in Section 14.3.8.

14.3-2-A **Site-Specific ITAAC**

STD COL 14.3-2-A This COL item is addressed in Section 14.3.9.

Appendix 14.3A Design Acceptance Criteria ITAAC Closure Process

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

14.3A.1 Design Acceptance Criteria ITAAC Closure Options

Replace the last two sentences of the second paragraph with the following.

NAPS COL 14.3A-1-1 Unit 3 is scheduled to be the first standard ESBWR plant licensed and will use the standard approach. A Design Acceptance Criteria ITAAC closure schedule will be provided for Unit 3 within one year after ESBWR design certification.

14.3A.5 COL Information

14.3A-1-1 **Establish a Schedule for Design Acceptance Criteria ITAAC Closure**

NAPS COL 14.3A-1-1 This COL item is addressed in Section 14.3A.1.

STD COL 14.2-1-A

Appendix 14AA Description of Initial Test Program Administration

14AA.1 Summary of Test Program and Objectives

14AA.1.1 Applicability

This appendix provides the requirements to be included in the Startup Administrative Manual (SAM), as discussed in DCD Sections 14.2.2.1 and 14.2.2.3. The information in and referenced in this appendix meets the ITP criteria of NUREG-0800 and is formatted to follow RG 1.206, Section C.I.14.2.

The ITP is applied to structures, systems, and components that perform the functions described in the RG 1.68 evaluation in Section 1.9. The ITP is also applied to other structures, systems, and components that meet any of the following criteria, even if not included in RG 1.68, Appendix A:

- Will be used for shutdown and cool down of the reactor under normal plant conditions, and for maintaining the reactor in a safe condition for an extended shutdown period.
- Will be used for shutdown and cool down of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions, and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions.
- Will be used to establish conformance with safety limits or limiting conditions for operation that will be included in the facility's Technical Specifications.
- Are classified as engineered safety features or will be relied on to support or ensure the operation of engineered safety features within design limits.
- Are assumed to function, or for which credit is taken, in the accident analysis of the facility, as described in the FSAR.
- Will be used to process, store, control, or limit the release of radioactive materials.

The SAM includes a list of the ESBWR structures, systems, and components to which the ITP is applied.

14AA.1.2 Phases of the Initial Test Program

The ITP (per RG 1.68) has the following five phases:

1. Preoperational Testing
2. Initial Fuel Loading and Pre-Criticality Tests
3. Initial Criticality
4. Low-Power Tests
5. Power Ascension Tests

These phases are described in further detail in DCD Section 14.2 and in Section 14.2, and are referred to collectively as Startup Tests.

14AA.1.3 Objectives of Preoperational and Startup Testing

Objectives of Preoperational Testing are in DCD Section 14.2.1.2. Objectives of Startup Testing are in DCD Section 14.2.1.3.

14AA.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 ESBWR plants is reviewed where available and the ITP modified as needed based on those lessons learned.

14AA.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.

14AA.2 Organization and Staffing

Administration of the ITP is governed by procedures in the SAM.

14AA.2.1 Organizational Description

The Plant Staff organization is described in Section 13.1. General preoperational responsibilities and a description of preoperational and startup testing are provided in Section 13AA.2. DCD Section 14.2.1.4 provides a description of the Startup Group organization.

The Startup Group has two internal groups: the Preoperational Test Group, which is responsible for conducting and documenting preoperational tests; and the Startup Test Group, which is responsible for conducting and documenting initial startup testing. Both groups consist of personnel drawn from various organizations such as plant staff, construction personnel, GEH, and other contractors, vendors and consultants.

The manager in charge of the Startup Group reports to the plant manager and has the qualifications of Preoperational Testing Supervisor as set forth in Table 13.1-201.

The Preoperational Test Group consists of Preoperational Testing Supervisors (i.e., NSSS, BOP, Electrical, and others, as required), each of whom reports to the manager in charge of the Startup Group. Preoperational Testing Engineers are assigned to this group and report to one of the Preoperational Testing Supervisors. Qualifications of Preoperational Testing Supervisors and Preoperational Testing Engineers are set forth in Table 13.1-201.

The Startup Test Group consists of Startup Testing Supervisors who report to the manager in charge of the Startup Group. Startup Test Engineers are assigned to this group and report directly to one of the Startup Testing Supervisors. Qualifications of Startup Testing Supervisors and Startup Test Engineers are set forth in Table 13.1-201. Figure 14AA-201 illustrates the organizational structure of the Startup Group.

14AA.2.2 **Responsibilities**

The manager in charge of Operations coordinates with the manager in charge of the Startup Group during the ITP to provide operations personnel to coordinate, support, and participate in preoperational testing. The manager in charge of Operations is a voting member of the Joint Test Group (JTG) and the Independent Review Body (IRB). The manager in charge of Operations is responsible for safe operation of the plant and ensuring tests are performed efficiently and effectively.

14AA.2.2.1 **Startup Group Manager**

The manager in charge of the Startup Group is responsible for:

- Staffing within the Startup Group.
- Developing procedures associated with ITP.
- Acting as Chairman of the JTG.
- Acting as an advisor to the IRB for all matters associated with startup testing.
- Managing contracts associated with the ITP.
- Coordinating with station and construction department heads for assignment of staff personnel to accomplish the test program objectives.

14AA.2.2.2 **GEH Resident Site Manager**

The GEH resident site manager is responsible for technical direction during the ITP. Qualifications of the GEH resident site manager are equivalent to the qualifications described in ANSI/ANS-3.1-1993 for a Preoperational Testing Supervisor. Specific responsibilities are:

- Acting as liaison with GEH on testing matters involving GEH-supplied equipment.
- Reviewing preoperational and startup test procedures, with emphasis on the GEH Nuclear Steam Supply System (NSSS).
- Assisting in data reduction, analysis, and evaluation for completed tests.
- Acting as a voting member of JTG.
- Providing administrative support and supervision to GEH onsite personnel involved in the test program.

14AA.2.2.3 **Vendor Site Representative**

A vendor site representative is responsible for technical direction during the preoperational phase of the test program. This position is filled as needed based on the scope of non-GEH supplied equipment that requires preoperational or startup testing. Specific responsibilities are:

- Acting as liaison with vendor on testing matters involving vendor supplied equipment.
- Reviewing preoperational tests with emphasis on vendor-supplied equipment.

- Assisting in data reduction, analysis, and evaluation for preoperational tests.
- Providing administrative support and supervision to vendor onsite personnel involved in the test program.

14AA.2.2.4 **Preoperational Testing Supervisor**

Preoperational Testing Supervisors are responsible for:

- Supervising the Preoperational Testing Engineers assigned to them.
- Coordinating and scheduling test preparation and test activities.
- Acting as voting member of JTG.
- Preparing, reviewing, and performing preoperational test procedures.
- Reviewing preoperational test results and making recommendations based on the results.
- Resolving deficiencies identified during preoperational inspection and test activities.
- Ensuring Preoperational Testing Engineers are not the same personnel who designed or are responsible for satisfactory performance of the system(s) or design features(s) being tested.

14AA.2.2.5 **Startup Testing Supervisor**

Startup Testing Supervisors are responsible for:

- Supervising the Startup Test Engineers assigned to them.
- Coordinating and scheduling test preparation and test activities.
- Coordinating and directing testing for their shift via the Operations Shift Supervisor for all initial startup testing.
- Assisting with preparing, reviewing, and performing startup test procedures.
- Reviewing, analyzing, and evaluating test results and data.
- Assisting in the resolution of deficiencies identified during startup testing activities.
- Coordinating with the planning and scheduling group for initial startup activities.
- Expediting testing progress as necessary to support project schedule.

- Ensuring Startup Test Engineers are not the same personnel who designed or are responsible for satisfactory performance of the system(s) or design features(s) being tested.

14AA.2.2.6 **Preoperational Testing Engineer**

Preoperational Testing Engineers are responsible for:

- Determining the nature and degree of testing required for assigned systems.
- Developing test activity milestones, target dates, and manpower requirements.
- Following construction progress to support test program requirements.
- Ensuring that the required detailed preoperational test procedures are available for review and approval.
- Identifying special or temporary equipment or services needed to support testing.
- Assuring test identification tagging and station tagging are implemented as necessary to support testing and turnover.
- Directing all participating groups during preparation for the execution of assigned tasks.
- Identifying and assisting in the resolution of deficiencies and problems found during the construction and testing of assigned systems and areas.
- Reviewing and evaluating test results and preparing test summaries.

14AA.2.2.7 **Startup Test Engineer**

Startup Test Engineers are responsible for:

- Preparing the required detailed startup test procedures and making them available for review and approval.
- Identifying special or temporary equipment or services needed to support testing.
- Directing all participating groups during preparation for the execution of assigned tasks.
- Identifying and assisting in the resolution of deficiencies found during the construction and testing of assigned systems.
- Reviewing and evaluating the test results and data.

- Coordinating with Operations during the execution of assigned tasks.
- Assisting in the supervision and inspection of Balance of Plant (BOP) work, reviewing installation and performance tests, and providing general advice on startup tests.
- Providing engineering support activities and services during startup turbine generator testing and Main Turbine Electro-Hydraulic Control (EHC) System testing.

14AA.2.2.8 **Joint Test Group**

The JTG is the primary review and approval organization during the preoperational test phase of the test program and is equivalent to the group referred to in DCD Section 14.2.1.4 as the Startup Controlling Group (SCG). The required JTG quorum is described in an administrative procedure in the SAM. The JTG is responsible for:

- Performing duties delineated in the SAM.
- Reviewing and approving all preoperational test procedures prior to testing.
- Reviewing and approving all major changes or revisions to JTG-approved test procedures.
- Reviewing and approving the overall preoperational test schedule and sequence.
- Reviewing and approving the results of preoperational tests.
- Recommending the disposition of test deficiencies.
- Recommending retests or supplemental tests as required.
- Determining system readiness for turnover to operations.

14AA.2.2.9 **Document Control Coordinator**

A document control coordinator reports to the manager in charge of the Startup Group and has the qualifications described in ANSI/ANS-3.I-1993 for a Startup Test Engineer. The document control coordinator is responsible for:

- Tracking test procedure changes.
- Reviewing, approving and tracking document changes (including drawings, vendor tech manuals, procedures, design changes, etc.).
- Verifying that the test schedules are up to date with regard to latest testing results.

- Processing final test packages through review and approval by the IRB.

14AA.2.2.10 Independent Review Body

Upon initial fuel load, the IRB assumes responsibility for tasks previously assigned to the JTG. The IRB is responsible for review of all procedures that require a regulatory evaluation under 10 CFR 50.59 and 10 CFR 72.48, as well as all tests and modifications that affect nuclear safety. The IRB is responsible for review of all startup test procedures. The organizational structure, functions, and responsibilities of IRB are described in Appendix 17AA. During the startup test phase, the IRB is advised by the manager in charge of the Startup Group and the GEH resident site manager. The IRB may be addressed by other titles such as Plant Operations Review Committee (PORC), On-site Safety Review Committee, or Plant Safety Review Committee (PSRC).

14AA.2.3 Operating and Technical Staff Participation

Operating and technical staff qualifications and experience requirements are:

- Plant staff qualification and experience requirements are in Chapter 13 and in this appendix.
- Contractor qualification and experience requirements are in this appendix and in approved contractor procedures.
- Vendor staff qualification and experience requirements are in this appendix and in approved vendor procedures.
- Architect Engineer staff qualification and experience requirements are in this appendix and in approved Architect Engineer procedures.

Plant staff participates in all phases of the ITP. Plant staff groups that participate include but are not limited to: Quality Assurance staff, Quality Control staff, Operations staff, Maintenance staff, Engineering staff, Planning, Scheduling and Outage planning staff, and Work Management staff, including work planners and schedulers. Operations staff participates in preoperational testing as part of gaining experience as described in Appendix 13BB. Refer to Figure 14AA-201 for identification of organizations that have one or more participants in the ITP.

14AA.2.4 Conflict of Interest

Members of the Startup Group responsible for formulating and conducting preoperational and startup tests are not the same individuals who designed or are responsible for satisfactory performance of the systems or design features being tested. This does not preclude members of the design organizations from participating in test activities.

14AA.2.5 Training Requirements

Training on the overall test program is conducted prior to scheduled preoperational and initial startup testing and as new employees are added to the test groups. A training program for each functional group in the organization is developed, with regard to the scheduled preoperational and startup testing, to ensure that the necessary plant staff is ready for commencement of the ITP. Additional discussion on staff training is found in Section 13.2, Appendices 13AA and 13BB, and Figure 13.1-202. The training program includes:

- Systems to be tested.
- Training by selected major equipment vendors (e.g., turbine, plant control).
- A review of test program administration.
- Content of test procedures, including acceptance criteria review.
- Test sequence.
- Test conduct and closure.

Specific Just-In-Time (JIT) training is conducted for operating crews and other personnel conducting certain startup tests. This JIT training may involve simulator training. Criteria to be considered when determining if JIT is used for a test include complexity of the test and plant response, such as tests that result in plant trips or other transients, or where they may occur. Accredited training program procedures describe the process for determining training topics to be conducted. The intention is to be as well prepared as possible to operate the plant safely.

14AA.3 Test Procedures

14AA.3.1 Procedure Development

DCD Sections 14.2.2.2 and 14.2.2.4 provide a general discussion concerning test procedure development and review. Section 13.5

provides detailed requirements for developing, reviewing, and scheduling administrative procedures.

Test procedures are written in accordance with a technical procedure writer's guide. This writer's guide provides for procedure validation. This validation may, in some cases, be through the use of an available plant reference simulator. The suitability of using the simulator to validate a test procedure is evaluated on a case by case basis. It may not be suitable, for example, to use the simulator to validate a procedure whose results are required to validate the simulator modeling.

Test procedures maximize the use of plant operating and maintenance procedures for test tasks. This can take the form of referencing a plant procedure to perform a task, or extracting the steps from the plant procedure for use in the preoperational and startup test procedures. This includes the use of emergency procedures for verifying appropriate emergency actions as described in DCD Section 14.2.5. Step-by-step instructions on how to conduct the applicable test are described and are coordinated with plant procedures wherever applicable in the test procedure. Test procedures contain cautions, warnings, and notes, using criteria established in the technical procedure writer's guide.

14AA.3.2 **Procedure Format and Content Requirements**

DCD Section 14.2.8.1 discusses technical information to be provided by GEH and others that form the technical basis for test procedure objectives and acceptance criteria.

Each preoperational and startup test procedure includes the following:

- Cover page

  The cover page provides approval signatures and effective dates (signatures may be maintained on file and may not appear on the cover page). The title and the unit designator water mark appear on the cover page. If the test is considered an infrequently performed test, this would appear on the cover page.

- Table of Contents

- Purpose and Test Objectives Section

  This section identifies the goal of the specific preoperational/startup test. This is established by stating those systems, subsystems, or components that are included in the test, and a series of summarized specific functions to be demonstrated during the test. Objectives of the test are stated. Many systems tests are intended to demonstrate that each of several initiating events produces one or more expected responses. These initiating events and the corresponding responses are identified.

- Description Section

  This section describes the power plateau, specific testing activities, operability impacts, systems affected, RPS trips, containment isolation, etc.

- Reference Section

  This section lists documents used to prepare or revise the pre-operational or startup test procedure and any documents used or referred to while performing the procedure.

- Special Tools and Equipment (Temporary Equipment Installations) Section

  This section lists test equipment and special tools not routinely carried, plus any unusual expendable items recommended to perform the procedure. This section also identifies temporary test equipment installations and test equipment instructions.

- Precautions and Limitations Section

  The test procedure highlights and clearly describes any and all precautions needed to ensure a reliable test or the safety of personnel or equipment including termination criteria for the test. Included are any special actions to be taken if the test is terminated at critical points in the test.

- Initial Conditions Section

  This section lists the plant conditions required to perform the test. Example: verify that the plant is operating at the 75 percent (+0, –5 percent) rod line. Each test of the operation of a system requires that certain other activities be performed first (e.g., completion of construction, construction and/or preliminary tests, inspections, and certain other preoperational tests or operations). Where appropriate, instructions are given pertaining to the system configuration, components that should or should not be operating, and other pertinent conditions that might affect the operation of the given system. The preoperational testing procedures include, as appropriate, these specific prerequisites, as illustrated by the following examples:

- •• Confirm that construction activities associated with the system have been completed and documented.
- •• Field inspections have been conducted to ensure that the equipment is ready for operation, including inspection for proper fabrication and cleanliness, checkout of wiring continuity and electrical protective devices, adjustment of settings on torque-limiting devices and calibration of instruments, verification that all instrument loops are operable and respond within required response times, and adjustment and settings of temperature controllers and limit switches.
- •• Confirm that test equipment is operable and properly calibrated.
- •• Confirm communications systems are functional for conducting the test.
- •• Access control is in place for personnel safety.
- •• Support or interface systems are functional.
- •• Confirm that prerequisite tests are conducted on individual components or subsystems to demonstrate that they meet their functional requirements.

Special environmental conditions are included in this section. Test procedures include provisions to test the equipment under environmental conditions as close as practical to those the equipment will experience in both normal and accident situations. However, many tests are conducted at ambient conditions due to the impracticality of achieving normal and accident conditions during preoperational testing.

- System Testing Section

This section provides detailed step-by-step instructions for each test. To the extent practical, the test procedures use approved normal plant operating procedures. Expected plant result is explicitly or implicitly stated in the instructions through verification or measurement steps. Each procedure requires necessary nonstandard arrangements to be restored to their normal status after the test is completed. Control measures such as jumper logs and check-off lists are specified.

Nonstandard bypasses, valve configurations, and instrument settings are identified and highlighted for return to normal. Nonstandard arrangements are carefully examined to ensure that temporary arrangements do not invalidate the test by interfering with proper testing of the as-built system.

- Data Collection Section

  The test procedures prescribe the data to be collected and the form in which the data are to be recorded. All entries are permanent. The administrative controls include an acceptable method for correcting an entry.

- Acceptance Criteria Section

  The test procedures clearly identify the criteria against which the success or failure of the test is judged, and account for measurement errors and uncertainties. In some cases, these are qualitative criteria. Where applicable, quantitative values with appropriate tolerances are designated as acceptance criteria. This section includes acceptance criteria for judgment of plant and system performance (as described in the applicable test specification). Those test criteria that show compliance with the Combined License ITAAC are identified in this section. When a test criterion for a preoperational test is not met, the Preoperational Testing Engineer documents the failure through the corrective action process and contacts the applicable preoperational test supervisor to determine actions to take (e.g., submitting a work request).

  For the startup test program, criteria are divided into three categories, depending on the significance of the parameter or function. The following paragraphs describe each kind of test criterion, and the actions to be taken by the Startup Test Engineer after an individual test criterion is not satisfied.

  •• Level I Criteria: Level I criteria relate to the values of process variables assigned in the design or analysis of the plant and component systems or associated equipment. Violation of these Level I criteria may have plant operational or plant safety implications. If a Level I test criterion is not satisfied, the plant must be placed in a suitable hold condition that is judged to be satisfactory to safety based on the results of prior testing. The Startup Test Engineer notifies the on-shift SRO, (who may declare the equipment inoperable), notifies the Startup Group manager/Startup Testing Supervisor, enters the condition in the corrective action program, and issues work requests as needed. Plant operating or test procedures or the Technical Specifications guide the decision on the direction to be taken. Startup tests compatible with this hold condition may be continued. Resolution of the problem must be documented and pursued by appropriate equipment adjustments or through engineering support personnel. Following resolution, the applicable test portion must be repeated to verify that the Level I requirement is ultimately satisfied. A description of the problem resolution shall be included in the report documenting the successful test.

- Level 2 Criteria: Level 2 criteria are specified as key plant performance requirements that are equipment design specification values or requirements for the measured response. The expected plant response is predicted by best estimate computer code and the desired trip avoidance margins. Level 2 failures that occur during tuning and system adjustment must be documented in the test report and following resolution, the applicable test portion must be repeated (retesting could occur at a higher power level with IRB approval) to verify that the Level 2 criterion requirement is satisfied. If a Level 2 criterion requirement is not satisfied after a reasonable effort, then the cognizant design and engineering organization shall document the results in the corrective action program with a full explanation of their recommendations. In order for the system as a whole to be acceptable, all Level 2 requirements must be satisfied or documentation provided that either modifies Level 2 requirements or changes specific design criteria.
- Level 3 Criteria: Level 3 criteria are associated with specifications on the expected or desired performance of individual control loop components. Meeting Level 3 criteria helps assure that overall system and plant response requirements are satisfied. Therefore, Level 3 criteria are to be viewed as highly desirable rather than required to be satisfied. Good engineering judgment is appropriate in the application of these rules. Since overall system performance is a mathematical function of its individual components, one component whose performance is slightly worse than specified can be accepted provided that a system adjustment elsewhere will

positively overcome the deficiency. Large deviations from Level 3 performance requirements are not allowable. If a Level 3 criterion requirement is not satisfied, the subject component or inner loop shall be analyzed closely. However, if all Level I and Level 2 criteria are satisfied, then it is not required to repeat the transient test to satisfy the Level 3 performance requirements. The occurrence of this Level 3 criterion failure shall be documented in the test report and entered into the corrective action program.

- Follow-on Task Section

  This section includes activities that must be performed to complete the test procedure.

  - Completion Notification

    This section is included to identify persons to be notified that the procedure has been satisfactorily or unsatisfactorily completed.

  - Procedure Reviews

    This section is included to specify required reviews and comments by various personnel.

  - Records Disposition

    Records disposition guidance is described in site-specific procedures.

  - Attachments

    Test procedure attachments provide supporting information and equations and evaluation methods to be used to analyze the obtained data. This attachment lists the signals to be recorded by the data collection equipment. Analysis and evaluation attachments outline the calculations to be performed and provide for an evaluation of the test.

    Upon completion of a given test, a preliminary evaluation is performed which 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.

  - Documentation of Test Results

    Records identify each observer and/or data recorder participating in the test, as well as the type of observation, identifying numbers of test or measuring equipment, results, acceptability, and action taken to correct any deficiencies. Administrative procedures specify the retention period of test result summaries, and require permanent retention of documented summaries and evaluations.

14AA.3.3 **Other Startup Test Procedures**

The need for special startup tests may arise due to unplanned conditions. The format and content requirements for preoperational and startup tests apply to these procedures.

14AA.3.4 **Test Procedure Changes**

If it is determined that procedure corrections (including changes in test sequence) are required before or during the conduct of the test, the test engineer suspends testing and notifies operations and test personnel of the required change. For all such corrections, the test engineer prepares and processes a procedure change request as delineated in a site-specific procedure for processing procedure changes. Revisions are classified into two categories based on the intent of the change. The intent of a procedure is the specific task or goal that is to be accomplished by the procedure.

Intent changes are changes to:

- Purpose.
- Initial conditions (or prerequisites).
- Acceptance criteria or tolerances.
- Scaling or setpoints.
- The method for meeting a commitment identified in the procedure.
- Step verification (independent or concurrent).
- System/component as-left condition(s).
- Reactivity management (changes that impact the operator's ability to monitor, control, or manipulate the reactor).
- Add or delete a subsection.
- Decrease personnel safety or fire protection effectiveness.
- Delete, relocate, or add a hold point.
- Caution or warning statements.
- Startup test procedure testing sequence.

Non-intent changes and revisions do not change the intent of the procedure (e.g., typographical error corrections). Review and approval requirements for procedure changes that do not change the intent are established in administrative procedures in the SAM.

Procedure changes that change the intent of the procedure receive the same level of review and approval as the original procedure. All test procedure intent changes will be revised against the following criteria (consistent with 10 CFR 50.59 and the design certification rule):

- Departure from Tier 1 information.
- Departure from Tier 2 information that significantly decreases the level of safety in accordance with 10 CFR 50.59(c)(1) and meets any one of eight criteria in 10 CFR 50.59(c)(2)(i) through (viii) or 10 CFR 52, Design Certification Appendix, Section VIII.B.5.b.
- Departure from Tier 2* information.
- Departure from Technical Specifications.

Preoperational test procedure intent changes involving Tier 1, Tier 2*, Technical Specifications, or Tier 2 that require a license amendment must be approved by the NRC prior to procedure completion and approval. Startup test procedure intent changes involving Tier 1, Tier 2*, Technical Specifications, or Tier 2 that require a license amendment must be approved by the NRC prior to procedure use. Timely notification of the NRC is made when procedures are changed that have been sent to the NRC.

14AA.4 Conduct of the Initial Test Program

14AA.4.1 Administrative Controls

ITP conduct is described in DCD Section 14.2.2.3. The SAM governs the ITP and will be 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.

14AA.4.2 Procedure Verification

Because 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 ensuring:

- 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 change(s) made since the procedure was originally approved for implementation in the areas of acceptance criteria, FSAR, Technical Specifications, piping changes, etc.
- Any new Operating Experience lessons learned (since preparation of the procedure) are incorporated into individual test procedures.

Procedures require signoff of 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 ensure that test status and equipment configuration are known 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 all deficiencies, discrepancies, exceptions, nonconformances and failures (collectively known as test exceptions) identified in the ITP. The corrective action documentation becomes part of the test documentation. GEH 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, GEH site representative or appropriate vendor, and JTG or IRB. 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.

14AA.4.3 **Work Control**

The Startup Group is responsible for preparing work requests when Construction organization assistance is required. Work requests are issued in accordance with a site-specific procedure 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.

14AA.4.4 **Measuring and Test Equipment (M&TE)**

During the preoperational test program, as well as the startup test program, most activities that lead to plant commercial operation involve design value verifications. M&TE used during these activities are properly controlled, calibrated, and adjusted at specified intervals to maintain accuracy within necessary limits. M&TE is governed by a site-specific procedure for control of M&TE. M&TE includes portable tools, gauges, instruments, and other measuring and testing devices not permanently installed, for example, startup test instruments prepared by the Preoperational Test Group as well as those provided by the Construction organization or by vendors.

A calibration program is implemented. For standard M&TE equipment, calibration procedures are prepared for each type of M&TE calibrated onsite. Calibration intervals are established for each item of M&TE. However, if the calibration requirement of a particular piece of M&TE is beyond the capabilities or resources of the plant staff, this M&TE is sent to an offsite certified calibration or testing agency. If special test equipment is necessary only for the ITP, the responsible vendor provides this equipment with the appropriate calibration documentation.

14AA.4.5 System Turnover

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), electrical schematic diagrams, 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.

14AA.4.6 Preoperational Testing

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.

14AA.4.7 **Startup Testing**

The startup testing program is based on increasing power in discrete steps. Major testing is performed at discrete power levels as described in DCD Section 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.

14AA.4.8 **Conduct of Modifications during the Initial Test Program**

Temporary modifications may be required to conduct certain tests. These modifications are documented in the test procedure. The test procedures contain restoration steps and retesting required to confirm satisfactory restoration to required configuration. Modifications may be performed by the Construction organization or the plant staff processes prior to NRC issuance of the 10 CFR 52.103g 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 ensure that preoperational testing and ITAAC remain valid. Modifications made following NRC issuance of the 10 CFR 52.103g finding are in accordance with plant staff processes and meet license conditions. Modifications that require change of ITAAC require NRC approval of the ITAAC change.

14AA.4.9 **Conduct of Maintenance during the Initial Test Program**

All corrective or preventive maintenance activities are reviewed to determine the scope of post-maintenance testing to be performed. Prior to NRC issuance of the 10 CFR 52.103g finding, post-maintenance testing is conducted and documented to ensure that associated preoperational testing and ITAAC remain valid. Maintenance performed following NRC issuance of the 10 CFR 52.103g finding is in accordance with plant staff processes and meets license conditions.

14AA.4.10 **Audits**

A comprehensive system of planned and periodic audits is carried out to verify compliance with the ITP in accordance with the Quality Assurance Program Description. Follow-up actions, including re-audit of deficient areas, are taken where indicated.

14AA.5 Review, Evaluation and Approval of Test Results

14AA.5.1 Review and Approval Responsibilities

The reactor vendor is responsible for reviewing and approving the results of all tests of supplied equipment. Architect Engineer representatives review and approve the results of all tests of supplied equipment. Other vendors' representatives review and approve the results of all tests of supplied equipment. Plant staff review and approval responsibilities are in Section 14AA.2. Final approval of individual test completion is by the plant manager after approval by the JTG or IRB.

14AA.5.2 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.

14AA.6 Test Records

Records retention requirements are in DCD Section 14.2.2.5 and in the Quality Assurance Program Description.

14AA.6.1 Startup Test Reports

Startup test reports are generated describing and summarizing the completion of tests performed during the ITP. A startup report is required per RG 1.16 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 to be reported by license conditions due to regulatory guide commitments.

14AA.7 Test Program Conformance with Regulatory Guides

Section 1.9 provides the evaluation of ITP conformance with the following RGs:

- RG 1.30, "Quality Assurance Requirements for Installation, Inspection and Testing of Instrumentation and Electrical Equipment (Safety Guide 30)."
- RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."
- RG 1.68, "Initial Test Program For Water-Cooled Nuclear Power Plants."
- RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release."
- RG 1.116, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems."
- RG 1.139, "Guidance for Residual Heat Removal."
- RG 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."
- RG 1.168, "Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants."

These RGs contain guidance that is included in the content of test procedures.

14AA.8 Utilization of Operating Experience

Administrative procedures provide methodologies for evaluating and initiating action for operating experience information (OE). DCD Section 14.2.4 describes the general use of operating experience by GEH in the development of the ITP.

14AA.8.1 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 Operating Experience Reports.
- INPO 06-001, "Operating Experience."

- INPO 06-001 Addendum.
- INPO 07-003, “INPO/ Utility Benchmarking for New Plant Deployment.”
- INPO 07-003 Addendum.
- INPO 86-023, “Guidelines for Nuclear Power Construction Projects.”
- INPO 94-005, “Standard Operation Support of Nuclear Plants.”
- INPO 94-03, “Review of Commercial Nuclear Power Industry Standardization Experience.”
- INPO Document AP-909, “Construction of Standard Nuclear Plants.”
- INPO NX-1067, “Browns Ferry Nuclear Plant Unit I Restart Operational Readiness Lessons Learned.”
- NRC RG 1.68, “Initial Test Programs For Water-Cooled Nuclear Power Plants.”
- SER 24-85, “Xenon Tilt Oscillation Following Control Rod Insertion Test (05-24-1985).”
- SER 29-86, “Inadvertent Rapid Cooldown and Depressurization During a Remote Shutdown Test (08-12-1986).”
- SOER 87-01, “Core Damaging Accident Following an Improperly Conducted Test (03-06-1987).”
- SOER 91-01, “Conduct of Infrequently Performed Tests or Evolutions.”

14AA.8.2 **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 ensure 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.
- Implement compensatory measures in accordance with guidance for infrequently performed tests or evolutions where appropriate.

14AA.8.3 Summary of Test Program Features Influenced by the Review

The conclusions from the preceding section were incorporated in Sections 14AA.3.1 and 14AA.3.2.

14AA.8.4 Use of OE during Test Procedure Preparation

Administrative procedures require review of recent internal and external operating experience when preparing test procedures.

14AA.8.5 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.

14AA.9 Trial Use of Plant Operating Procedures and Emergency Procedures

14AA.9.1 Use of Plant Procedures during Initial Test Program

Whenever practical, plant procedures are used to perform system and component operation during the conduct of a test.

14AA.9.2 Operator Training and Participation during Certain Initial Tests (TMI Action Plan Item I.G.1, NUREG-0737)

The objectives of operator participation are to increase the capability of shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and off-normal events is conducted.

The major objective of TMI Action Plan Task I.G.1 was to use the preoperational and startup test programs as a training exercise for operating crews. NUREG-0933 contains a discussion of the proposed actions and the conclusions made. NUREG-0800, Section 14 was revised to address the original issue of this action item. NUREG-0933 discusses three anticipated operational occurrences applicable to the ESBWR. These are pressure controller failed high, pressure controller failed low, and stuck-open safety/relief valve. These events are addressed in the abnormal operating procedures. Operators receive training on them as part of their initial training. Operators participate in preoperational and startup testing.

Operators are trained on the specifics of the ITP schedule, administrative requirements and tests. Specific JIT training is conducted for selected startup tests.

The ITP may result in discovery of acceptable plant or system response differing from 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.

14AA.10 Initial Fuel Loading and Initial Criticality

14AA.10.1 Prerequisites for Fuel Loading

- Preoperational tests are completed or justification is documented and approved for test exceptions and tests that have not been performed.
- All ITAAC are complete and the NRC has issued 10 CFR 52.103g declaration.
- Technical Specifications required for fuel load are met.
- License Conditions are met to allow fuel load.
- Licensed operators are stationed in the control room and for supervision of core alterations.
- Composition, duties, and emergency procedure responsibilities of the fuel handling crew are specified.
- Persons are technically qualified in accordance with plant procedures.
- Radiation monitors, nuclear instrumentation, manual initiation, and other devices are tested and verified to be operable to actuate the building evacuation alarm and ventilation control.
- Status of each system required for fuel loading is specified.
- Inspections of fuel and control rods are complete and all identified issues with installed fuel and control rods are resolved.

- Nuclear instruments are calibrated, operable and properly located (source-fuel-detector geometry). One operating channel has audible indication or annunciation in the control room.
- A response check of nuclear instruments to a neutron source consistent with the Technical Specifications surveillance frequency for source range nuclear instruments in the refueling mode is complete.
- Required status of containment is specified and met.
- Required status of the reactor vessel is specified and met. Components are either in place or out of the vessel, as specified, to be capable of receiving fuel.
- Vessel water level is established, and the minimum level for fuel loading and unloading is specified.
- The standby liquid control system is operable.
- Fuel handling equipment is confirmed functional and operable through surveillance and other tests, including dry runs.
- The status of protection systems, interlocks, mode switches, alarms, and radiation protection equipment is prescribed and verified.
- Water quality is established within prescribed limits.

14AA.10.2 **Fuel Loading Procedure Details**

The fuel loading procedure includes instructions or information for the following areas:

- Loading sequence and pattern for fuel, control rods, and other components, with guidance regarding fuel addition increments so that the reactivity worth of added individual fuel assemblies becomes less as the core is assembled.
- Maintenance of a display for indicating the status of the core and fuel pool, as well as appropriate records of core loading.
- Proper seating and orientation of fuel and components (the procedure specifies a visual check of each assembly in each core position).
- Functional testing of each control rod immediately following fuel loading.
- Nuclear instrumentation and neutron source requirements for monitoring subcritical multiplication, including source or detector relocation and normalization of count rate after relocation.

- Flux monitoring, including counting times and frequencies and rules for plotting inverse multiplication and interpreting plots (the counting period for count rates is specified, and an inverse multiplication plot is maintained).
- The expected subcritical multiplication behavior.
- The minimum shutdown margin is proved periodically during loading and at the completion of loading. Shutdown margin verifications do not involve planned approach to criticality using nonstandard rod patterns or with operational interlocks bypassed.
- Actions (especially those pertaining to flux monitoring) for periods when fuel loading is interrupted.
- Maintenance of continuous voice communication between the control room and loading station.
- Minimum crew required to load fuel (the procedure requires the presence of at least two persons at any location where fuel handling is taking place, and a senior reactor operator with no other concurrent duties be in charge).
- Crew work time limits per 10 CFR 26 are in effect.
- Approvals required for changing the procedure.

14AA.10.3 **Fuel Loading Procedure Limitations and Actions**

The fuel loading procedure includes the following limits and instructions:

- Established criteria for stopping fuel loading. Some circumstances that might warrant this are unexpected subcritical multiplication behavior, loss of communications between the control room and fuel loading station, inoperable source-range detector, and inoperability of the emergency boration system.
- Established criteria for emergency boron injection.
- Established criteria for containment evacuation.
- Actions to be performed in the event of fuel damage.
- Actions to be performed and/or approvals to be obtained before routine loading may resume after one of the above limitations has been reached or invoked.

14AA.10.4 Initial Criticality Procedure Requirements

The format and content requirements for preoperational tests apply to the initial criticality procedure. Plant operations are in accordance with plant operating procedures to the maximum extent possible. This procedure includes steps to ensure that the startup proceeds in a deliberate and orderly manner, changes in reactivity are continuously monitored, and inverse multiplication plots are maintained and interpreted.

The initial criticality procedure includes the following requirements:

- A critical rod position is predicted so that any anomalies may be noted and evaluated.
- All systems needed for startup are aligned and in proper operation.
- The standby liquid control system is operable.
- Procedural, license and Technical Specification requirements are met for initial criticality.
- Nuclear instruments are calibrated. A neutron count rate (of at least one-half count per second) should register on neutron monitoring channels before the startup begins, and the signal-to-noise ratio should be known to be greater than two. A conservative startup rate limit (no shorter than approximately a 30-second period) is established.

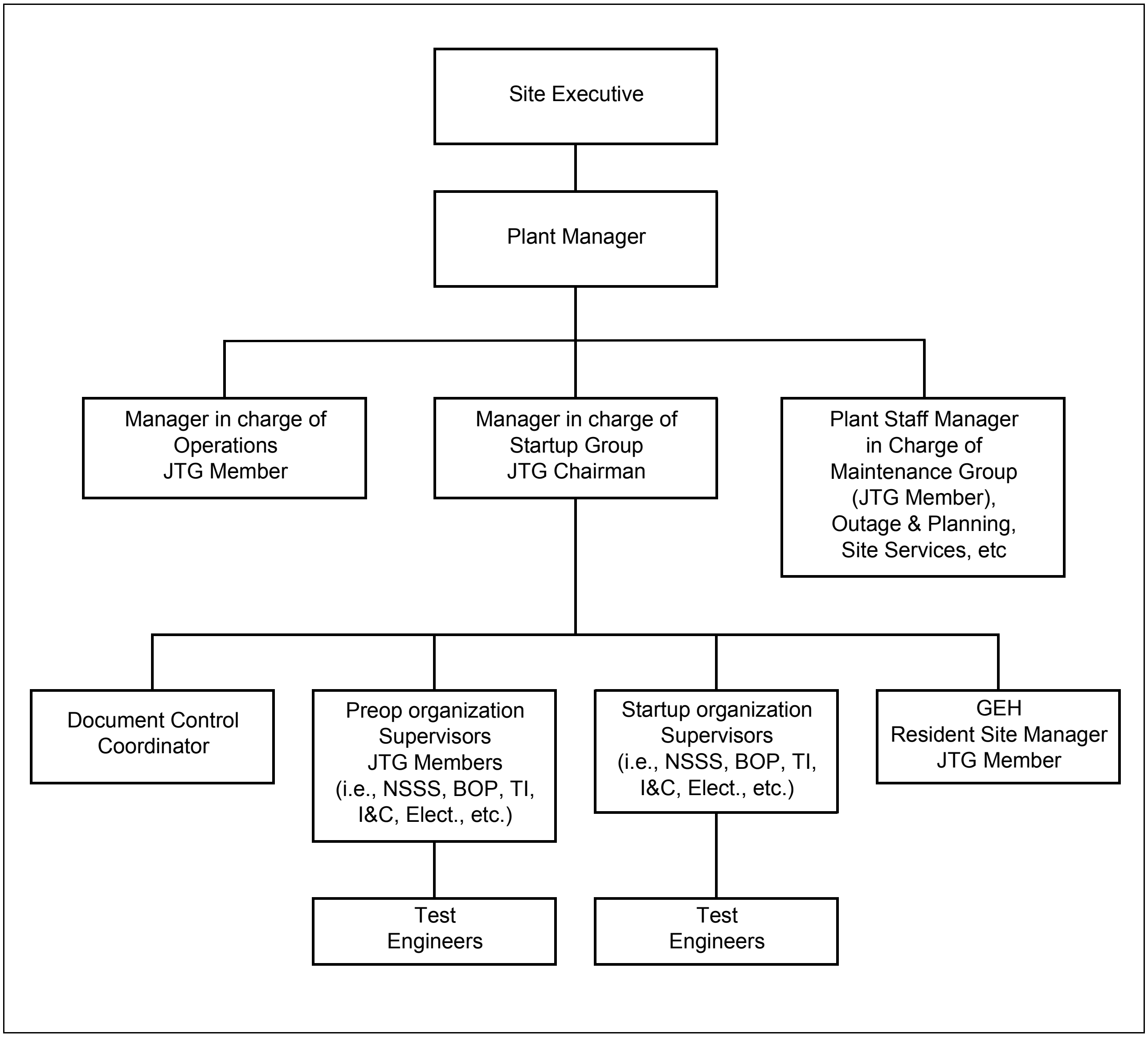
14AA.11 Plant Procedure Development Schedule

The milestone schedule for developing plant operating procedures is presented in Table 13.5-202 and discussed in Section 13.5.2.1. 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.

14AA.12 Individual Test Descriptions

Individual test descriptions can be found in DCD Section 14.2.8 and in Section 14.2.9.

Figure 14AA-201 Preoperational and Startup Test Organization (Typical)



Chapter 15 Safety Analyses

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

15.3 Analysis of Infrequent Events

15.3.10.5 Radiological Consequences

Add the following sentence at the end of this section.

STD SUP 15.3-1

In addition, procedures discuss the use of nuclear instrumentation to aid in detecting a possible mislocated fuel bundle after fueling operations.

NAPS SUP 15.3-2

15.6 ESP Information

NAPS ESP VAR 2.0-6

SSAR Chapter 15 is incorporated by reference except that information related to the ESBWR is replaced by DCD Chapter 15.

Chapter 16 Technical Specifications

16.0 Introduction

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

STD SUP 16.0-1 The Technical Specifications and the Technical Specification Bases are maintained as separate documents.

16.0.1 COL Information

16.0-1-A COL Applicant Bracketed Items

STD COL 16.0-1-A This COL item is addressed in the Technical Specifications and Technical Specification Bases.

16.0-2-H COL Holder Bracketed Items

STD COL 16.0-1-H This COL item is addressed in the Technical Specifications and Technical Specification Bases.

Chapter 17 Quality Assurance

17.0 Introduction

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

Add the following after the last paragraph.

NAPS SUP 17.0-1

The QAPD applicable to the COL licensee is described in Section 17.5. The licensee's QAPD describes the basis of the program, its scope of activities, and the control of work performed by suppliers.

17.1 Quality Assurance During Design

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

Add the following after the first paragraph.

NAPS SUP 17.1-1

Quality Assurance (QA) applied during the preparation of the ESPA is described in SSAR Chapter 17, which is incorporated by reference.

NAPS SUP 17.1-2

QA applied during COL application preparation and site specific design activities is addressed in Section 17.5.

17.2 Quality Assurance During Construction and Operations

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

Replace the first paragraph with the following.

NAPS COL 17.2-1-A
NAPS COL 17.2-2-A

The licensee's Quality Assurance Program in place during the construction and operations phases, including adapting the design to specific plant implementation, is described in Section 17.5.

17.2.1 COL Information

17.2-1-A QA Program for the Construction and Operations Phases

NAPS COL 17.2-1-A

This COL Item is addressed in Section 17.2.

17.2-2-A QA Program for Design Activities

NAPS COL 17.2-2-A

This COL Item is addressed in Section 17.2.

17.3 Quality Assurance Program Description

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

Replace the first and second sentences with the following.

NAPS COL 17.3-1-A The Quality Assurance Program Document applicable to the licensee is described in Section 17.5.

17.3.1 COL Information

17.3-1-A Quality Assurance Program Document

NAPS COL 17.3-1-A This COL Item is addressed in Section 17.3.

17.4 Reliability Assurance Program During Design Phase

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

17.4.1 Introduction

Replace the third paragraph and subsequent bulleted list with the following.

STD COL 17.4-1-H The objectives of reliability assurance during the operations phase are integrated into the Quality Assurance Program (Section 17.5), the Maintenance Rule (MR) Program (Section 17.6), and other operational programs. Specific reliability assurance activities are addressed within operational programs (e.g., maintenance rule, surveillance testing, inservice testing, inservice inspection, and quality assurance) and the maintenance programs.

The MR Program incorporates the following aspects of operational reliability assurance (refer to Section 17.6):

- Use of PRA importance measures, the expert panel process, and deterministic methods to determine the list of risk-significant SSCs
- Evaluation and maintenance of the reliability of risk-significant SSCs
- Monitoring the effectiveness of maintenance activities needed for operational reliability assurance
- Classifying, initially, as high-safety-significant, all SSCs that are in the scope of the design reliability assurance program (D-RAP), or applying expert panel review for any exceptions

- Use of historical data and industry operating experience on equipment performance as available
- Use of specific criteria to establish the level of performance or condition being maintained for SSCs within the scope of the MR Program; and use of monitoring to identify declining trends between surveillances and to minimize the likelihood of undetected performance or condition degradation to unacceptable levels, to the extent possible
- Use of maintenance programs to determine the nature and frequency of maintenance activities to be performed on plant equipment, including SSCs within the scope of the MR Program

17.4.6 SSC Identification/Prioritization

Add the following new paragraph at the end of this section.

STD COL 17.4-1-H

The list of risk-significant SSCs will be confirmed via ITAAC (see DCD Tier 1 Table 3.6-1).

17.4.9 Operational Reliability Assurance Activities

Replace the second paragraph with the following.

STD COL 17.4-1-H

Refer to Section 17.4.1 for the implementation of reliability assurance during the operations phase.

17.4.10 Owner/Operator's Reliability Assurance Program

Replace the fifth bullet with the following.

STD COL 17.4-1-H

- **MR Program:** The MR Program is described in Section 17.6.

Replace the last sentence in this section with the following.

Refer to Section 17.4.1 for the implementation of reliability assurance activities.

17.4.13 COL Information

17.4-1-H Operation Reliability Assurance Activities

STD COL 17.4-1-H

This COL Item is addressed in Sections 17.4.1, 17.4.6, 17.4.9, 17.4.10, and 17.6.

NAPS COL 17.3-1-A

17.5 Quality Assurance Program Description - Design Certification, Early Site Permits, and New License Applicants

QA applied to the DC activities is described in DCD Section 17.1.

QA applied during the preparation of the ESP application is described in SSAR Chapter 17.

NAPS SUP 17.5-2

QA applied to safety-related activities performed prior to start of construction (e.g., site investigation, design and safety analysis, early procurements) is described in the Dominion Nuclear Facility QAPD (Reference 17.5-201) topical report for the Dominion operating nuclear plants as supplemented by COL Project procedures.

NAPS COL 17.2-1-A
NAPS COL 17.2-2-A

QA applied to activities to adapt the design to specific plant implementation, construction, and operations is addressed in the Dominion QAPD (Appendix 17AA). The QAPD is based on NEI 06-14. (Reference 17.5-202)

The implementation milestones for the Operational Quality Assurance Program are provided in Section 13.4.

17.5.1 References

17.5-201 DOM-QA-1, Dominion Nuclear Facility Quality Assurance Program Description.

17.5-202 Nuclear Energy Institute, “Quality Assurance Program Description,” NEI 06-14.

STD COL 17.4-1-H

17.6 Maintenance Rule Program

NEI 07-02, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52” (Reference 17.6-201) is incorporated by reference with the following supplemental information:

STD SUP 17.6-1

The text of the template provided in NEI 07-02 is generically numbered as “17.X.” When the template is incorporated by reference into this section, numbering is changed from “17.X” to “17.6.”

STD SUP 17.6-3

17.6.1.1 Maintenance Rule Scoping per 10 CFR 50.65(b)

In Paragraph 17.6.1.1.b, replace "(DRAP - see FSAR Section 17.Y)" with the following.

(See Section 17.4)

17.6.3 Maintenance Rule Program Relationship with Reliability Assurance Activities

Replace with the following.

STD SUP 17.6-2

Reliability during the operations phase is assured through the implementation of operational programs, i.e., the MR program (Section 17.6), the Quality Assurance Program (Section 17.5), the Inservice Inspection Program (Sections 5.2.4 and 6.6, and DCD Section 3.8.1.7.3), and the Inservice Testing Program (Sections 3.9.6 and 3.9.3.7.1(3)e), as well as the Technical Specifications Surveillance Requirements (Chapter 16) and maintenance programs.

17.6.6 References

17.6-201 Nuclear Energy Institute, "Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52," NEI 07-02.

NAPS SUP 17.5-3

Appendix 17AA North Anna Power Station Unit 3 Quality Assurance Program Description

North Anna Unit 3 Quality Assurance Program Description

Topical Report
DOM-QA-2

Revision 0

Policy
Quality Assurance During Construction and Operation

Dominion Virginia Power (Dominion) shall design, procure, construct and operate the North Anna Unit 3 nuclear plant in a manner that will ensure the health and safety of the public and workers. These activities shall be performed in compliance with the requirements of the Code of Federal Regulations (CFR), the applicable Nuclear Regulatory Commission (NRC) Facility Operating Licenses, and applicable laws and regulations of the state and local governments.

The Dominion North Anna Unit 3 Quality Assurance Program (QAP) is the Quality Assurance Program Description (QAPD) provided in this document and the associated implementing documents. Together they provide for control of Dominion activities that affect the quality of safety-related nuclear plant structures, systems, and components (SSCs) and include all planned and systematic activities necessary to provide adequate confidence that such SSCs will perform satisfactorily in service. The QAPD may also be applied to certain equipment and activities that are not safety-related, but support safe plant operations, or where other NRC guidance establishes program requirements.

The QAPD is the top-level policy document that establishes the manner in which quality is to be achieved and presents Dominion's overall philosophy regarding achievement and assurance of quality. Implementing documents assign more detailed responsibilities and requirements and define the organizational interfaces involved in conducting activities within the scope of the QAP. Compliance with the QAPD and implementing documents is mandatory for personnel directly or indirectly associated with implementation of the Dominion North Anna Unit 3 QAP.

Signed *Signature on file*

David A. Christian

Senior Vice President Nuclear Operations & Chief Nuclear Officer

LIST OF EFFECTIVE SECTIONS

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Part I Introduction

Section 1 General

Dominion's North Anna Unit 3 Quality Assurance Program Description (QAPD) is the top-level document that establishes the quality assurance policy and assigns major functional responsibilities for combined construction and operating license (COL) activities conducted by or for Dominion. The QAPD describes the methods and establishes quality assurance (QA) and administrative control requirements that meet 10 CFR 50, Appendix B, and 10 CFR 52. The QAPD is based on the requirements and recommendations of ASME NQA-1-1994, "Quality Assurance Requirements for Nuclear Facility Applications," Parts I, II and III, as specified in this document.

The QAP is defined by the NRC-approved regulatory document that describes the QA elements (i.e., the QAPD), along with the associated implementing documents. Procedures and instructions that control North Anna Unit 3 activities will be developed prior to commencement of those activities. Dominion policies establish high-level responsibilities and authority for carrying out important administrative functions. Procedures establish practices for certain activities that are common to all Dominion nuclear business unit organizations performing those activities so that the activity is controlled and carried out in a manner that meets QAPD requirements. Procedures specific to a site, organization, or group establish detailed implementation requirements and methods, and may be used to implement policies or be unique to particular functions or work activities.

1.1 Scope/Applicability

The QAPD applies to COL, construction/pre-operation and operations, activities affecting the quality and performance of safety-related structures, systems, and components, including, but not limited to:

Designing	Cleaning
Siting	Testing
Training	Inspecting
Constructing	Preoperational activities (including ITAAC*)
Procuring	Startup
Receiving	Operating
Storing	Maintaining
Handling	Repairing
Shipping	Refueling
Erecting	Modifying
Installing	Decommissioning
Fabricating	

* ITAAC are those Inspections, Tests, Analyses, and Acceptance Criteria the applicant must satisfy as determined by the Commission in accordance with 10 CFR Part 52.

Safety-related SSCs, under the control of the QAPD, are identified by design documents. The technical aspects of these items are considered when determining program applicability, including, as appropriate, the item's design safety function. The QAPD may be applied to certain activities where regulations other than 10 CFR 50 and 10 CFR 52 establish QA requirements for activities within their scope.

The policy of Dominion is to assure a high degree of availability and reliability of the nuclear plant while ensuring the health and safety of its workers and the public. To this end, selected elements of the QAPD are also applied to certain equipment and activities that are not safety-related, but support safe, economic, and reliable plant operations, or where other NRC guidance establishes quality assurance requirements. Implementing documents establish program element applicability.

The definitions provided in ASME NQA-1-1994, Part 1, Section 1.4, apply to select terms as used in this document.

Part II QAPD Details

Section 1 Organization

This section describes the Dominion organizational structure, functional responsibilities, levels of authority and interfaces for establishing, executing, and verifying QAPD implementation. The organizational structure includes corporate support and onsite functions for North Anna Unit 3 including interface responsibilities for multiple organizations that perform quality-related functions. Implementing documents assign more specific responsibilities and duties, and define the organizational interfaces involved in conducting activities and duties within the scope of the QAPD. Management gives careful consideration to the timing, extent and effects of organizational structure changes.

Dominion Nuclear Oversight Senior Manager is responsible to size the Quality Assurance organization commensurate with the duties and responsibilities assigned.

The following sections describe the reporting relationships, functional responsibilities and authorities for organizations implementing and supporting the North Anna Unit 3 QA Program. Titles used herein are generic functional descriptions. Administrative documents are maintained to relate the generic titles to the Dominion specific titles. The Dominion organizations for the North Anna Unit 3 construction and operations phases are shown in Figures II-1 and II-2, respectively.

1.1 Chief Nuclear Officer

The Chief Nuclear Officer (CNO) has overall responsibility and authority for implementing all activities associated with the safe and reliable design, construction, operation, and decommissioning of Dominion's nuclear facilities. The CNO establishes the North Anna Unit 3 quality assurance policy and provides guidance regarding its implementation. The CNO has delegated the responsibility and authority for approval of the QAPD to the senior manager of the group responsible for nuclear oversight. The CNO has the authority to resolve disputes related to implementation of the QAPD for which resolution is not achieved at lower levels within the organization. There are six functional organizations reporting to the CNO that affect the safety of the nuclear facilities: Nuclear Development, Nuclear Plant Construction, Nuclear Oversight, Operations, Engineering Services, and Support Services.

1.2 Nuclear Development

An executive management position is responsible for the development of new nuclear power plants. This includes activities associated with new nuclear plant engineering, analysis, design, procurement, pre-construction preparation, preparing applications, and obtaining permits and licenses for potential construction. Where implementation of any or all of these functions is delegated to organizations outside Dominion, procedures require the establishment of interface documents including defining lines of communication and authorities as appropriate for the delegated functions. However, this executive management position retains responsibility for the scope and effective implementation of the quality assurance program for those functions.

1.2.1 Manager Engineering

The functional manager for Engineering is responsible for the conduct of reviews of design developed by the suppliers, preservice testing activities during the preoperational and startup period. This group is responsible procurement of items and services for the new plant. The new plant engineering group is also responsible for document control and collection and preservation of records.

1.2.1.1 Engineering Design Service

Dominion has contracted with General Electric - Hitachi (GEH) and Bechtel to perform design activities in support of construction and to adapt the ESBWR design to the North Anna site. Dominion has delegated the responsibility of establishing and executing quality assurance measures for this design activity to the respective companies in accordance with their approved quality assurance programs. Dominion has also contracted with GEH to provide safety-related long lead-time components under the GEH quality assurance program. Dominion reviews and audits the suppliers on a regular basis to ensure conformance with the quality requirements.

1.2.1.2 Preservice Testing and Inspection

Preservice testing and inspection duties and responsibilities include the establishment of the programs to ensure required preservice inspections and tests for the plant are identified, performed and documented. This group is responsible for the incorporation of inspection and test information into the plant inservice inspection and test program and the submittal of required reports.

1.2.1.3 Plant Engineering

Plant engineering duties and responsibilities include establishing and implementing a process for selection and review of design documents produced for the North Anna Unit 3 project. The selection is based on the item's safety significance, complexity of design, standardization, state of the art, and/or similarity to other proven designs. This group is also responsible for providing input to and technical review of Dominion purchase documents for services and items related to the project, including any necessary interface with existing Dominion systems and groups. This group is responsible for the review and approval of the resolution of nonconforming items with a disposition of "accept-as-is" or "repair', i.e., where the action does not fully restore the item to the supplier's design, manufacturing and/or testing specifications, requirements of the purchase document, or purchaser approved documents.

1.2.1.4 Document Control

This group is responsible for control of Dominion documents, such as instructions, procedures, and drawings. This group is also responsible for the collection and preservation of quality assurance records. This group interfaces with existing Dominion processes and personnel in performing these duties.

1.2.2 Plant Manager

The plant manager's group is responsible for developing a trained and qualified staff for licensed operating activities. They accept control of SSCs turned over to Dominion for operation. Functional activities include operation, control, maintenance and start-up testing of the SSCs. This group is responsible for transitioning into the operating phase organization as described in Chapter 13 of the FSAR.

1.2.2.1 Operations

This group is responsible for operation and control of SSCs that have been turned over to Dominion. This group controls the starting and stopping of components, isolation for maintenance, and assists in the conduct of preoperational and startup tests as necessary. This group ensures sufficient staff are trained and licensed as necessary to progress into the operations phase of the plant.

1.2.2.2 Startup Testing

The startup test group is responsible for the development of procedures, planning, scheduling, and executing startup tests in accordance with the procedures. Qualified test personnel are used to review and approve the procedures. Qualified individuals are designated to review and approve test results.

1.2.2.3 Maintenance

The maintenance group is responsible for developing, scheduling, and performing periodic maintenance, including associated special processes, necessary to ensure the quality of installed SSCs that have been turned over to Dominion. This group assists in the performance of preoperational and startup testing, as necessary, and maintains control of measuring and test equipment for use by Dominion.

1.2.2.4 Training

The training group is responsible to develop and implement a systematic approach to training for operations and maintenance personnel. This group ensures fidelity of the training simulator with the constructed plant. The training group is responsible for achieving and maintaining accreditation of the training program. This group interfaces with the corporate training group for support as needed.

1.2.3 Organizational Effectiveness

The manager of the organizational effectiveness group is responsible for the corrective action program and the construction experience program.

1.2.4 Licensing and Regulatory Interface

The manager of the licensing and regulatory interface group is responsible for corresponding with the NRC on regulatory matters including the combined construction and operating license, other permits, and the completion of ITAAC requirements.

1.3 Nuclear Plant Construction

An executive management position is responsible for the construction of new nuclear power plants. This position assists in establishing contracts, provides oversight and coordination of construction contractors, and manages Dominion's cost and schedules. Suppliers will be used to perform the majority of engineering, procurement, and construction activities. The suppliers will be delegated the responsibility for achieving and assuring quality of the SSCs, however, Dominion retains the overall responsibility for quality.

1.3.1 Manager Construction

The manager of construction is responsible for interfacing with contractors for coordination of the overall construction effort to keep the project moving, controlling cost and schedule while maintaining quality of work. The manager construction ensures a process is developed and implemented to identify and resolve construction interferences so that changes are reflected back to the design and as-built configuration of the plant.

1.3.1.1 Scheduling

The scheduling personnel provide oversight and coordination of the overall project schedule development and implementation.

1.3.1.2 Cost

The cost personnel are responsible for oversight of cost controls for the construction phase of the project.

1.3.1.3 Procurement

The procurement personnel are responsible for the development and issuance of Dominion procurement documents for the project. They may interface with existing Dominion systems and personnel in the performance of this activity.

1.3.1.4 Fire Protection and Industrial Safety

The fire protection and industrial safety personnel are responsible for the development and implementation of fire protection measures and the industrial safety program during construction. This includes monitoring of supplier performance in these areas.

1.3.2 Principal Design and Construction Supplier(s)

Dominion will procure the services of one or more suppliers to develop and implement the North Anna Unit 3 construction project. This will include the activities of detailed construction engineering, procuring items, and the construction and installation of SSCs for the facility. Designation of the entities and their responsibilities will be added to this organization description as the suppliers are identified. Dominion will delegate to these suppliers the duties of and responsibility for establishing and executing a QA program for the design, procurement, manufacture, fabrication, installation, inspection, and testing of SSCs for the North Anna Unit 3 facility.

1.4 Nuclear Oversight

A senior management position is responsible for the verification of effective Dominion and Supplier QA program development, documentation, and implementation. This position is independent of cost and scheduling concerns associated with construction, operations, maintenance, modification, and decommissioning activities for performing quality assurance program verification. Where implementation of any or all of these functions is delegated to Suppliers, procedures require the establishment of interface documents including defining lines of communication and authorities as appropriate for the delegated functions. However, this senior management position retains responsibility for the scope and effective implementation of the quality assurance program for those functions. This management position has the necessary authority and responsibility for verifying quality achievement; identifying quality problems, recommending solutions and verifying implementation of the solutions; and escalating quality problems to higher management levels. This position has the authority to suspend unsatisfactory work and control further processing or installation of non-conforming materials. The authority to stop work delegated to Nuclear Oversight personnel is delineated in procedures.

1.4.1 Nuclear Development Phase (Construction)

Nuclear Oversight is responsible for QA oversight of the North Anna Unit 3 project. The oversight includes activities in development of the license application, design, procurement, construction, and related activities that affect the quality of SSCs.

1.4.1.1 QA Program Development

This group is responsible for development and maintenance of the QAPD. This group is responsible for verification of the development of the construction QA program through review of and concurrence in quality-related procedures for design, construction, and installation. This group also performs audits of the effectiveness of the QA program implementation within the Dominion new plant development organization.

1.4.1.2 Site QA/QC

This group is responsible for quality oversight of supplier conducted activities at the North Anna construction site through a system of planned audits, surveillances, and inspections as appropriate to the activity and based on the importance of the item or activity to the safety of the plant. This group is responsible for performance of inspections for Dominion activities on SSCs that have been turned over to Dominion for operation.

1.4.1.3 Supplier QA/QC

This group is responsible for quality oversight of suppliers and is performed through a system of audits, surveillances, and inspections as appropriate to the activity and based on the importance of the item or activity to the safety of the plant. This oversight is conducted at Dominion's and Suppliers' facilities. In performance of the oversight, this group will interface with Dominion's existing systems and groups for qualifying suppliers and performing verification activities.

1.4.2 Operations Phase

Nuclear Oversight is responsible for the evaluation of Suppliers' quality programs through a system of external audits, evaluations, and reviews of Supplier performance in accordance with quality assurance requirements. A list of approved Suppliers is maintained. Nuclear Oversight is responsible for assuring Dominion compliance with the QAPD through administration of a comprehensive and systematic internal audit program.

Nuclear Oversight is responsible for developing and maintaining an appropriate quality verification inspection program where not provided for in the facility construction or operating organization functions.

1.4.2.1 Facility Oversight

A management position is responsible for the effective performance of Nuclear Oversight activities. This position performs independent assessment of facility operations related to quality and safety with lines of communication to the executive management position responsible for facility operations.

1.4.2.1.1 Quality Control Inspection

The Quality Control Inspection group plans and conducts inspections of operating facility maintenance and modification activities to ensure quality in accordance with the requirements of the QA program. The Quality Control Inspectors report through this functional organization while performing maintenance and modification inspections for the operating facility.

1.5 Operations (Operations Phase)

An executive management position is responsible for overall operating activities of Dominion's nuclear facilities. This executive is responsible for implementing the quality assurance program during operating activities, including related decommissioning activities.

1.5.1 Facility Operations

An executive management position is responsible for operations of their assigned Dominion nuclear facilities. The necessary responsibility and authority for the management and direction of all activities related to the safe and efficient operation and decommissioning has been delegated by the senior executives. This responsibility includes ensuring quality through implementation of the QAPD in all the activities related to operation such as maintenance, testing, start-up and shutdown, refueling, fuel storage, and modification.

1.5.1.1 Facility Operations and Maintenance

A senior management position is responsible for safe operations and maintenance of the nuclear facilities including those activities necessary for safe storage and handling of spent nuclear fuel during decommissioning. The position responsibilities include: directing the operations, maintenance, planning, and site services groups; implementing facility modifications; and maintaining compliance with requirements of the operating license, Technical Specifications, and applicable federal, state, and local laws, regulations, and codes.

1.5.1.1.1 **Operations**

Operations is responsible for operating the facility in accordance with the applicable license, including those in a decommissioning phase that still contain nuclear fuel. Overall facility operation is directed by a management position responsible for Operations activities.

Operations activities include monitoring and controlling day-to-day operation of the nuclear facility; responding to alarms; manipulating facility equipment; coordinating facility operations to manage work such as maintenance, testing, and modifications; and moving nuclear fuel. The Operations organization contains supervision and staff for shift operations, including shift managers, unit supervisors, licensed control room operators, and non-licensed operators. Operations is also responsible for the shift technical advisor function. Operations is also responsible for oversight of fire protection measures.

1.5.1.1.2 **Maintenance**

Maintenance is responsible for directing and coordinating facility maintenance activities including on-line maintenance, installation, maintenance, alterations, adjustment and calibration, replacement and repair of plant electrical and mechanical equipment, and instruments and controls. The responsibilities include performance of surveillances required by Technical Specifications, establishing standards and frequency of calibration for instrumentation and control devices, and ensuring instrumentation and related testing equipment are properly used, inspected and maintained.

1.5.1.1.3 **Outage and Planning**

Outage & Planning is responsible for planning and scheduling online-maintenance and outage activities.

1.5.1.1.4 **Site Services**

Site Services is responsible for facility project support, including project construction and project controls.

1.5.1.2 **Safety and Licensing**

A senior management position is responsible for ensuring that facility safety and licensing requirements are implemented. This position is responsible for directing and coordinating radiological protection and assessment of nuclear safety issues at the facility, including independent review functions through the independent review body (IRB). The responsibilities also include managing licensing activities; interfacing with corporate management on operating experience and licensing issues, managing facility procedures, and administering the facility environmental compliance program. This position is independent of cost and scheduling concerns associated with operations, maintenance, and modification activities. This position has the authority to suspend unsatisfactory work and control further processing or installation of non-conforming materials. The authority to stop work delegated to quality control inspection personnel is delineated in procedures.

1.5.1.2.1 **Organizational Effectiveness**

Organizational Effectiveness is responsible for the corrective action program and the operating experience program.

1.5.1.2.2 **Radiological Protection and Chemistry**

Radiological Protection & Chemistry carries out health physics and chemistry functions and maintains sufficient organizational freedom and independence from operating pressures as required by the facility Technical Specifications. A qualified supervisor or manager is assigned to fulfill the radiological protection manager position described in Section 2.6 of the QAPD. The radiological protection responsibilities include scheduling and conducting radiological surveys, contamination sample collection, determining contamination levels, assigning work restrictions through radiation work permits, administering the personnel monitoring program, and maintaining required records in accordance with federal and state codes. The chemistry responsibilities include maintaining primary and secondary plant chemistry in accordance with established program requirements.

1.5.1.2.3 **Procedures**

The Procedures group is responsible for ensuring that procedures are prepared in accordance with applicable regulatory requirements, industry quality standards, and the QAPD.

1.5.1.2.4 **Licensing**

The licensing group is responsible for corresponding with the NRC on license related matters and supporting arrangements for NRC inspections.

1.5.1.2.5 Quality Control

The quality control group is responsible for the development and implementation of a QC program for the operating unit. The responsibilities include the planning of inspections and a process for assigning inspections to qualified individuals. This group is responsible for oversight of the training, qualification and certification of inspection personnel.

1.6 Engineering Services (Operations Phase)

An executive management position is responsible for the engineering functions supporting design and construction activities and long-term nuclear operations. These are accomplished through nuclear engineering, projects, nuclear analysis and fuel, information technology, and document control and records management groups. Responsibilities include system level implementation of the requirements established by the QAPD for the nuclear facilities and facility specific engineering and technical support required for day-to-day operations. Where implementation of any or all of these functions is delegated to organizations outside Dominion, procedures require the establishment of interface documents including defining lines of communication and authorities as appropriate for the delegated functions. However, this executive management position retains responsibility for the scope and effective implementation of the quality assurance program for those functions.

1.6.1 Nuclear Engineering

A senior management position is responsible for design engineering functions and supporting activities. Such as independent design checks and reviews, developing and maintaining engineering programs, including those for nondestructive examination (NDE), and the facility inservice inspection and test (ISI/IST) programs; configuration management including design and configuration control, and developing and revising facility drawings; and engineering technical support at the operating facilities.

1.6.1.1 Design Engineering

Design engineering is responsible for managing engineering resources providing day-to-day technical support for facility operations. The functions include engineering and technical support at a system and component level to ensure optimum design basis performance, system reliability, and optimum component performance and reliability. Support is also provided in developing and implementing testing programs, tracking and scheduling test performance, and evaluating test results. The test programs include inservice inspections, Technical Specification surveillances, post-modification and post-maintenance testing, and nondestructive examinations.

1.6.1.2 Project Engineering

Project engineering is responsible for the implementation of large projects for the nuclear facilities on behalf of Dominion. Implementation includes development of the detailed scope, estimate, schedule, cost, design, procurement, construction, testing, and closeout of each project. Project engineering focuses on defined projects separate from ongoing routine engineering projects.

1.6.1.3 Engineering Programs

Engineering programs is responsible for providing support in classifying SSCs, maintaining the design control program, developing and implementing the inservice inspection and test programs, and ensuring the design basis for the facility is maintained.

1.6.1.3.1 Document Control and Records Management

Document control and records management groups are assigned responsibility to ensure controlled documents (such as manuals, instructions, procedures, and drawings) and facility records are maintained in accordance with applicable regulatory requirements, industry quality standards, and the QAPD.

1.6.1.3.2 Fire Protection Engineer

The fire protection engineer is responsible for maintaining the fire protection design basis and assisting with the resolution of problems related to fire protection at the site.

1.6.2 Nuclear Analysis and Fuel

A senior management position is responsible for activities related to safety and management of nuclear fuel. Nuclear Analysis and Fuel (NAF) is responsible for engineering activities, evaluation, and analysis of: core design, fuel and reactor performance, probabilistic risk assessment, spent fuel storage, and radiological effects. NAF provides reactor-engineering support for the operating power stations. NAF is responsible for nuclear fuel procurement, assurance of nuclear fuel quality through surveillances and inspections at Dominion and supplier facilities, and special nuclear material accountability. This position has the authority to control further processing or installation of nonconforming materials. The authority delegated to inspection and surveillance personnel is delineated in procedures. NAF is also responsible for providing engineering oversight of dry cask spent fuel storage system fabrication, including approval of nonconformance disposition.

1.6.3 Information Technology

A senior management position is responsible for direction and support of information technology for the nuclear organizations and facilities. Responsibilities include: network infrastructure maintenance and upgrade, network and application security, network operations; automation strategy, application development and support, automation training; development and maintenance of the software control program; and oversight, maintenance, and repair of the Emergency Response Facility Computer System.

1.7 Support Services (Operations Phase)

An executive management position is responsible to provide licensing, fire protection, security, emergency preparedness, training, and procurement support services to the Nuclear Organization. Where implementation of any or all of these functions is delegated to organizations outside Dominion, procedures require the establishment of interface documents including defining lines of communication and authorities as appropriate for the delegated functions. However, this executive management position retains responsibility for the scope and effective implementation of the quality assurance program for those functions.

1.7.1 Licensing and Operations Support

A senior management position is responsible for providing regulatory compliance and licensing support through NRC communications, maintaining and acquiring licenses required for continued and extended operations and providing operations, chemistry and health physics support.

1.7.2 Protection Services and Emergency Preparedness

A senior management position is responsible for providing nuclear facility security, and overall management of Nuclear Emergency Preparedness activities.

1.7.2.1 Protection Services

Protection Services is responsible for facility protective services, including physical security, nuclear facility access programs, and fitness for duty programs. Protection Services is also responsible for industrial safety and loss prevention including oversight of fire protection measures.

1.7.2.2 Emergency Preparedness

Emergency Preparedness is responsible for development and maintenance of Dominion radiological emergency plans and coordination with required off-site radiological emergency response groups for the nuclear facilities. This includes managing the overall scheduling and coordination of emergency plan testing, training and exercises with federal, state, and local agencies, and working with corporate and facility personnel to ensure emergency plans meet all the requirements and commitments.

1.7.3 Training

A senior management position is responsible for the training of personnel who operate or support the nuclear facilities. Training responsibilities include: determining the need for training based on information provided by the various groups, developing performance-based training programs, implementing training programs to support employee and facility needs, and evaluating training programs. Certain functional groups may be assigned responsibility for the development and conduct of their own training programs provided these groups are not required to have a systems approach to training under 10 CFR 50.120.

1.7.4 Supply Chain Management

A senior management position is responsible for material management, purchasing, procurement engineering, Supplier surveillance functions, and source and receipt inspection. This position has the authority to control further processing or installation of nonconforming materials. This authority is delegated to inspection and surveillance personnel as delineated in procedures.

1.8 Authority to Stop Work

Quality assurance and inspection personnel have the authority, and the responsibility, to stop work in progress which is not being done in accordance with approved procedures or where safety or SSC integrity may be jeopardized. This extends to offsite work performed by suppliers that furnish safety-related materials and services to Dominion.

1.9 Quality Assurance Organizational Independence

For the COL construction activities, independence shall be maintained between the organization or organizations performing the checking (quality assurance and control) functions and the organizations performing the functions. This provision Is not applicable to design review/verification.

1.10 NQA-1-1994 Commitment

In establishing its organizational structure, Dominion commits to compliance with NQA-1-1994, Basic Requirement 1 and Supplement 1S-1.

Figure II-1 Construction Organization

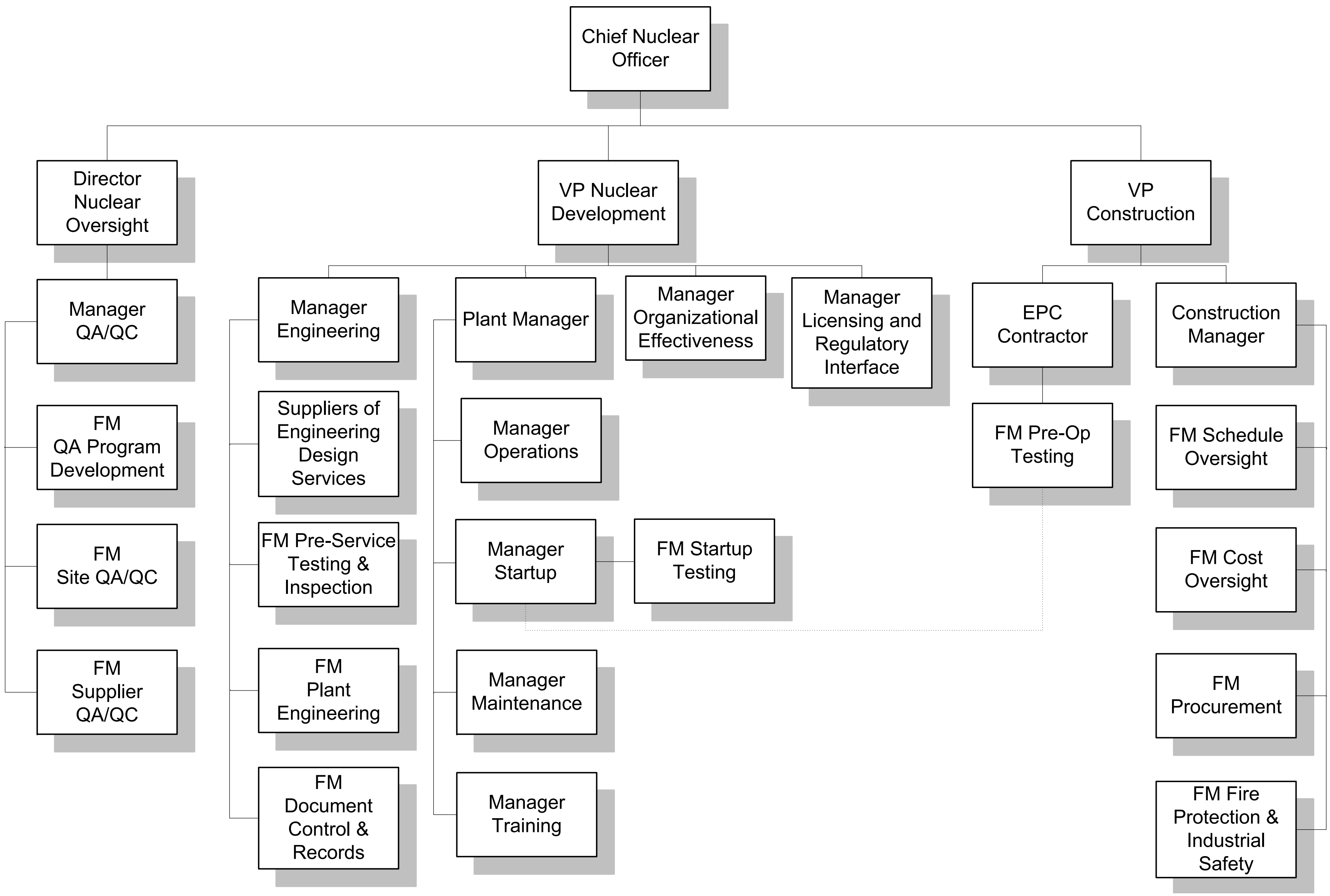
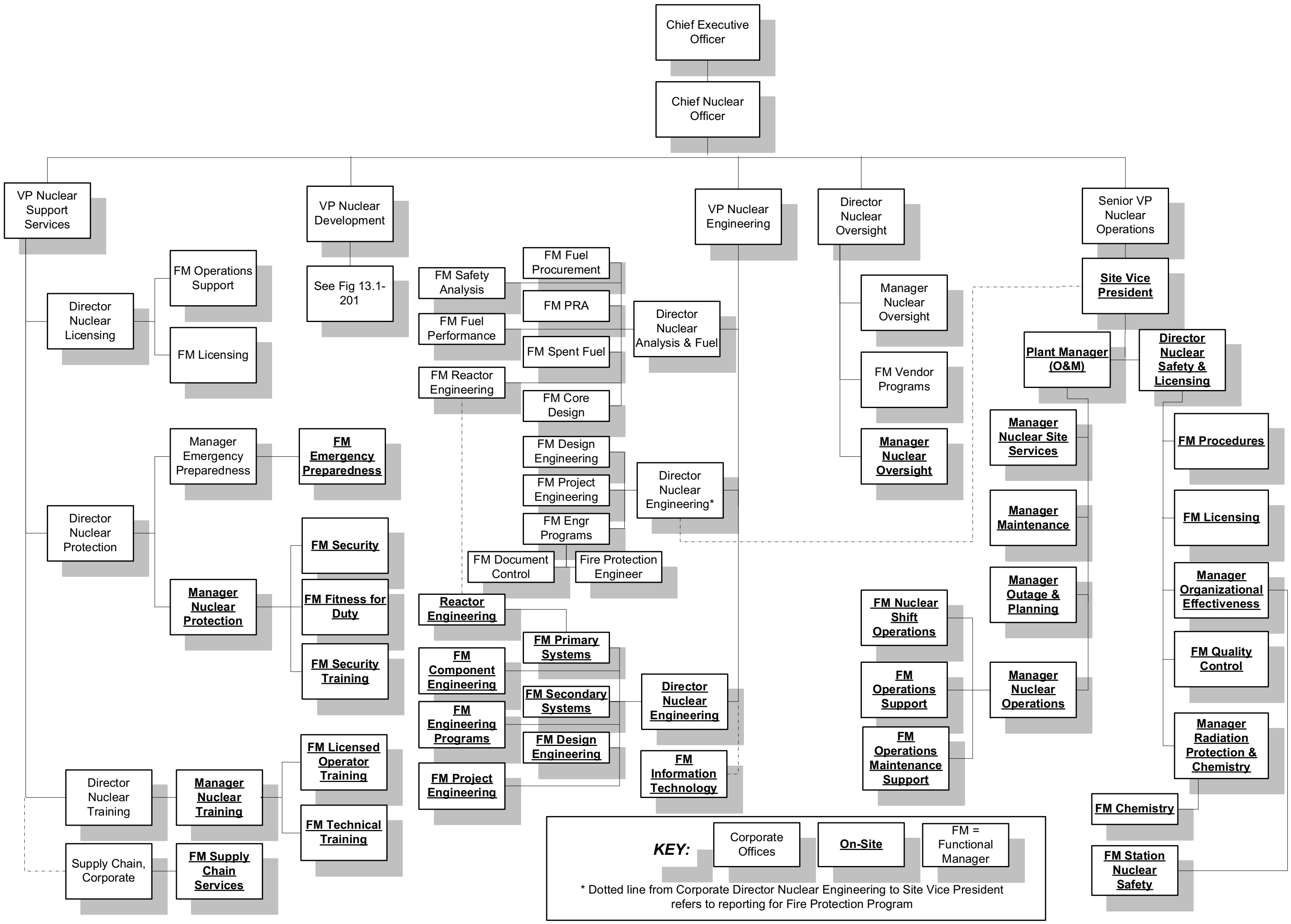


Figure II-2 Operating Organization



Section 2 Quality Assurance Program

Dominion has established the necessary measures and governing procedures to implement the QAP as described in the QAPD. Dominion is committed to implementing the QAP in all aspects of work that are important to the safety of the nuclear plant as described and to the extent delineated in the QAPD. Further, Dominion ensures through the systematic process described herein that its suppliers of safety-related equipment or services meet the applicable requirements of 10 CFR 50, Appendix B. Senior management is regularly apprised of the adequacy of implementation of the QAP through the audit functions described in Part II, Section 18.

The objective of the QAP is to assure that the North Anna Unit 3 nuclear generating plant is designed, constructed, and operated in accordance with governing regulations and license requirements. The program is based on the requirements of ASME NQA-1-1994, “Quality Assurance Requirements for Nuclear Facility Applications,” as further described in this document. The QAP applies to those quality-related activities that involve the functions of safety-related SSCs associated with the design (excluding Design Certification activities), fabrication, construction, and testing of the facility SSC, and to the managerial and administrative controls used to assure safe operations. Examples of COL safety-related activities include, but are not limited to, site-specific engineering related to safety-related SSCs, site geotechnical investigations, site engineering analysis, seismic analysis, and meteorological analysis. A list or system that identifies SSCs and activities to which this program applies is maintained at the appropriate facility. The Design Certification Document is used as the basis for this list or system. Cost and scheduling functions do not prevent proper implementation of the QAP.

As described in Part III of the QAPD, specific program controls are applied to nonsafety-related SSCs, for which 10 CFR 50, Appendix B, is not applicable, that are significant contributors to plant safety. The specific program controls consistent with applicable sections of the QAPD are applied to those items in a selected manner, targeted at those characteristics or critical attributes that render the SSC a significant contributor to plant safety.

Delegated responsibilities may be performed under a supplier’s or principal contractor’s QA Program, provided that the supplier or principal contractor has been approved as a supplier in accordance with the QAP. Periodic audits and assessments of supplier QA programs are performed to assure compliance with the supplier’s or principal contractor’s QAPD and implementing procedures. In addition, routine interfaces with supplier’s personnel provide added assurance that quality expectations are met.

For the COL application, the QAPD applies to those North Anna Unit 3 and Dominion activities that can affect either directly or indirectly the safety-related site characteristics or analysis of those characteristics. In addition, the QAPD applies to engineering activities that are used to characterize the site or analyze that characterization.

New nuclear plant construction will be the responsibility of Dominion's North Anna Unit 3 organization. Detailed engineering specifications and construction procedures will be developed to implement the QAPD and EPC QA programs prior to commencement of construction (COL) activities. Examples of Limited Work Authorization (LWA) activities that could impact safety-related SSCs include impacts of construction to existing facilities and for construction of new plants, the interface between nonsafety-related and safety-related SSCs, and the placement of seismically-designed backfill.

In general, the program requirements specified herein are detailed in implementing procedures that are either Dominion/North Anna Unit 3 implementing procedures, or supplier implementing procedures governed by a supplier quality assurance program.

A grace period of 90 days may be applied to provisions that are required to be performed on a periodic basis unless otherwise noted. Annual evaluations and audits that must be performed on a triennial basis are examples where the 90 day grace period could be applied. The grace period does not allow the "clock" for a particular activity to be reset forward. The "clock" for an activity is reset backwards by performing the activity early. Audits schedules are based on the month in which the audit starts.

2.1 Responsibilities

Personnel who work directly or indirectly for Dominion are responsible for achieving acceptable quality in the work covered by the QAPD. This includes those activities delineated in Part I, Section 1.1. Dominion personnel performing verification activities are responsible for verifying the achievement of acceptable quality. Activities governed by the QAPD are performed as directed by documented instructions, procedures, and drawings that are of a detail appropriate for the activity's complexity and effect on safety. Instructions, procedures, and drawings specify quantitative or qualitative acceptance criteria, as applicable or appropriate for the activity, and verification is against these criteria. Provisions are established to designate or identify the proper documents to be used in an activity, and to ascertain that such documents are being used. The North Anna Unit 3 nuclear oversight manager is responsible to verify that processes and procedures comply with the QAPD and other applicable requirements, that such processes or procedures are implemented, and that management appropriately ensures compliance.

2.2 Delegation of Work

Dominion retains and exercises the responsibility for the scope and implementation of an effective QAP. Positions identified in Part II, Section 1, may delegate all or part of the activities of planning, establishing, and implementing the program for which they are responsible to others, but retain the responsibility for the program's effectiveness. Decisions affecting safety are made at the level appropriate for its nature and effect, and with any necessary technical advice or review.

2.3 Site-Specific Safety-Related Design Basis Activities

Site-specific safety-related design basis activities are defined as those activities, including sampling, testing, data collection, and supporting engineering calculations and reports, that will be used to determine the bounding physical parameters of the site. Appropriate quality assurance measures are applied.

2.4 Periodic Review of the Quality Assurance Program

Management of those organizations implementing the QA program, or portions thereof, assess the adequacy of that part of the program for which they are responsible to assure its effective implementation at least once each year or at least once during the life of the activity, whichever is shorter. However, the period for assessing QA programs during the operations phase may be extended to once every two years.

2.5 Issuance and Revision to Quality Assurance Program

Administrative control of the QAPD will be in accordance with 10 CFR 50.55(f) and 10 CFR 50.54(a), as appropriate. Changes to the QAPD are evaluated by the nuclear oversight manager to ensure that such changes do not degrade previously approved quality assurance controls specified in the QAPD. This document shall be revised as appropriate to incorporate additional QA commitments, that may be established during the COL application development process. New revisions to the document will be reviewed, at a minimum, by the Dominion manager responsible for North Anna Unit 3 nuclear oversight and approved by the senior manager responsible for Dominion's nuclear oversight group.

Regulations require that the FSAR include, among other things, the managerial and administrative controls to be used to assure safe operation, including a discussion of how the applicable requirements of Appendix B will be satisfied. In order to comply with this requirement, the FSAR references the QAPD and, as a result, the requirements of 10 CFR 50.54(a) are satisfied by and apply to the QAPD.

2.6 Personnel Qualifications

Personnel assigned to implement elements of the QAPD shall be capable of performing their assigned tasks. To this end, Dominion establishes and maintains formal indoctrination and training programs for personnel performing, verifying, or managing activities within the scope of the QAPD to assure that suitable proficiency is achieved and maintained. Plant and support staff minimum qualification requirements are as delineated in the unit Technical Specifications. Other qualification requirements may be established but will not reduce those required by Technical Specifications. Sufficient managerial depth is provided to cover absences of incumbents. When required by code, regulation, or standard, specific qualification and selection of personnel is conducted in accordance with those requirements as established in the applicable Dominion procedures. Indoctrination includes the administrative and technical objectives, requirements of the applicable codes and standards, and the QAPD elements to be employed. Training for positions identified in 10 CFR 50.120 is accomplished according to programs accredited by the National Nuclear Accrediting Board of the National Academy of Nuclear Training that implement a systematic approach to training. Records of personnel training and qualification are maintained.

The minimum qualifications of the senior management position for Dominion's nuclear oversight group and the management position for North Anna Unit 3 nuclear oversight are that each holds an engineering or related science degree and a minimum of four years of related experience including two years of nuclear power plant experience, one year of supervisory or management experience, and one year of the experience is in performing quality verification activities. Special requirements shall include management and supervisory skills and experience or training in leadership, interpersonal communication, management responsibilities, motivation of personnel, problem analysis and decision making, and administrative policies and procedures. Individuals who do not possess these formal education and minimum experience requirements should not be eliminated automatically when other factors provide sufficient demonstration of their abilities. These other factors are evaluated on a case-by-case basis and approved and documented by senior management.

The minimum qualifications of the individuals responsible for planning, implementing, and maintaining the QAPD are that each has a high school diploma or equivalent and has a minimum of one year of related experience. Individuals who do not possess these formal education and minimum experience requirements should not be eliminated automatically when other factors provide sufficient demonstration of their abilities. These other factors are evaluated on a case-by-case basis and approved and documented by senior management.

2.7 Independent Review

Activities occurring during the operational phase shall be independently reviewed on a periodic basis. The independent review program shall be functional prior to initial core loading. The independent review function performs the following:

- Reviews proposed changes to the facility as described in the safety analysis report (SAR). The Independent Review Body (IRB) also verifies that changes do not adversely affect safety and if a technical specification change or NRC review is required.
- Reviews proposed tests and experiments not described in the SAR. Changes to proposed tests and experiments not described in the SAR that do require a technical specification change must be reviewed by the IRB prior to NRC submittal and implementation.

- Reviews proposed technical specification changes and license amendments relating to nuclear safety prior to NRC submittal and implementation, except in those cases where the change is identical to a previously approved change.
- Reviews violations, deviations, and events that are required to be reported to the NRC. This review includes the results of investigations and recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.
- Reviews any matter related to nuclear safety that is requested by the Site Executive or any IRB member.
- Reviews corrective actions for significant conditions adverse to quality.
- Reviews the adequacy of the audit program every 24 months.

A group may function as an independent review body (IRB). In discharging its review responsibilities, the IRB keeps safety considerations paramount when opposed to cost or schedule considerations. One or more organizational units may collectively perform this function.

IRB reviews are supplemented as follows:

- A qualified person, independent of the preparer, reviews proposed changes in the procedures as described in the SAR prior to implementation of the change to determine if a technical specification change or NRC approval is required.
- Audits of selected changes in the procedures described in the SAR are performed to verify that procedure reviews and revision controls are effectively implemented.
- Competent individual(s) or group(s) other than those who performed the original design but who may be from the same organization verify that changes to the facility do not result in a loss of adequate design or safety margins.

The results of IRB reviews of matters involving the safe operation of the facility are periodically independently reviewed. This review is intended to support management in identifying and resolving issues potentially affecting safe plant operation. This review supplements the existing corrective action programs and audits.

- The review is performed by a team consisting of personnel with experience and competence in the activities being reviewed, but independent from cost and schedule considerations and from the organizations responsible for those activities.
- The review is supplemented by outside consultants or organizations as necessary to ensure the team has the requisite expertise and competence.
- Results of the review are documented and reported to responsible management.

- Management periodically consider issues they determine warrant special attention, such as deficient plant programs, declining performance trends, employee concerns, or other issues related to safe plant operations and determine what issues warrant the review.
- Management determines the scheduling and scope of review and the composition of the team performing the review.

2.8 NQA-1-1994 Commitment/Exceptions

In establishing qualification and training programs, Dominion commits to compliance with NQA-1-1994, Basic Requirement 2 and Supplements 2S-1, 2S-2, 2S-3 and 2S-4, with the following clarifications and exceptions:

- NQA-1-1994, Supplement 2S-1
  - Supplement 2S-1 will include use of the guidance provided in Appendix 2A-1 the same as if it were part of the Supplement. The following two alternatives may be applied to the implementation of this Supplement and Appendix:
    - (1) In lieu of being certified as Level I, II, or III in accordance with NQA-1-1994, personnel that perform independent quality verification inspections, examinations, measurements, or tests of material, products, or activities will be required to possess qualifications equal to or better than those required for performing the task being verified; and the verification is within the skills of these personnel and/or is addressed by procedures. These individuals will not be responsible for the planning of quality verification inspections and tests (i.e., establishing hold points and acceptance criteria in procedures, and determining who will be responsible for performing the inspections), evaluating inspection training programs, nor certifying inspection personnel.
    - (2) A qualified engineer may be used to plan inspections, evaluate the capabilities of an inspector, or evaluate the training program for inspectors. For the purpose of these functions, a qualified engineer is one who has a baccalaureate in engineering in a discipline related to the inspection activity (such as electrical, mechanical, civil) and has a minimum of five years engineering work experience with at least two years of this experience related to nuclear facilities.
- NQA-1-1994, Supplement 2S-2
  - In lieu of Supplement 2S-2, for qualification of nondestructive examination personnel, North Anna Unit 3 will follow the applicable standard cited in the version(s) of Section III and Section XI of the ASME Boiler and Pressure Vessel Code approved by the NRC for use at the North Anna Unit 3 site.
- NQA-1-1994, Supplement 2S-3

- The requirement that prospective Lead Auditors have participated in a minimum of five audits in the previous three years is replaced by the following, “The prospective lead auditor shall demonstrate his/her ability to properly implement the audit process, as implemented by North Anna Unit 3, to effectively lead an audit team, and to effectively organize and report results, including participation in at least one nuclear audit within the year preceding the date of qualification.”

Section 3 Design Control

Dominion has established and implements a process to control the design, design changes, and temporary modifications (e.g.,temporary bypass lines, electrical jumpers and lifted wires, and temporary setpoints) of items that are subject to the provisions of the QAPD. The design process includes provisions to control design inputs, outputs, changes, interfaces, records, and organizational interfaces within Dominion and with suppliers. These provisions assure that design inputs (such as design bases and the performance, regulatory, quality, and quality verification requirements) are correctly translated into design outputs (such as analyses, specifications, drawings, procedures, and instructions) so that the final design output can be related to the design input in sufficient detail to permit verification. Design change processes and the division of responsibilities for design-related activities are detailed in North Anna Unit 3 and supplier procedures. The design control program includes interface controls necessary to control the development, verification, approval, release, status, distribution, and revision of design inputs and outputs. Design changes and disposition of nonconforming items as “use as is” or “repair” are reviewed and approved by the North Anna Unit 3 design organization or by other organizations so authorized by Dominion.

Design documents are reviewed by individuals knowledgeable in QA to ensure the documents contain the necessary QA requirements.

3.1 Design Verification

Dominion design processes provide for design verification to ensure that items and activities subject to the provisions of the QAPD are suitable for their intended application, consistent with their effect on safety. Design changes are subjected to these controls, which include verification measures commensurate with those applied to original plant design.

Design verifications are performed by competent individuals or groups other than those who performed the original design but who may be from the same organization. The verifier shall not have taken part in the selection of design inputs, the selection of design considerations, or the selection of a singular design approach, as applicable. This verification may be performed by the originator’s supervisor provided the supervisor did not specify a singular design approach, rule out certain design considerations, and did not establish the design inputs used in the design, or if the supervisor is the only individual in the organization competent to perform the verification. If the verification is performed by the originator’s supervisor, the justification of the need is documented and approved in advance by management.

The extent of the design verification required is a function of the importance to safety of the item under consideration, the complexity of the design, the degree of standardization, the state-of-the-art, and the similarity with previously proven designs. This includes design inputs, design outputs, and design changes. Design verification procedures are established and implemented to assure that an appropriate verification method is used, the appropriate design parameters to be verified are chosen, the acceptance criteria are identified, and the verification is satisfactorily accomplished and documented. Verification methods may include, but are not limited to, design reviews, alternative calculations and qualification testing. Testing used to verify the acceptability of a specific design feature demonstrates acceptable performance under conditions that simulate the most adverse design conditions expected for the item's intended use.

North Anna Unit 3 normally completes design verification activities before the design outputs are used by other organizations for design work, and before they are used to support other activities such as procurement, manufacture, or construction. When such timing cannot be achieved, the design verification is completed before relying on the item to perform its intended design or safety function.

3.2 Design Records

Dominion maintains records sufficient to provide evidence that the design was properly accomplished. These records include the final design output and any revisions thereto, as well as record of the important design steps (e.g., calculations, analyses and computer programs) and the sources of input that support the final output.

Plant design drawings reflect the properly reviewed and approved configuration of the plant.

3.3 Computer Application and Digital Equipment Software

The QAPD governs the development, procurement, testing, maintenance, and use of computer application and digital equipment software when used in safety-related applications and designated nonsafety-related applications. Dominion and suppliers are responsible for developing, approving, and issuing procedures, as necessary, to control the use of such computer application and digital equipment software. The procedures require that the application software be assigned a proper quality classification and that the associated quality requirements be consistent with this classification. Each application software and revision thereto is documented and approved by the code manager as delineated in the software control procedures. The QAPD is also applicable to the administrative functions associated with the maintenance and security of computer hardware where such functions are considered essential in order to comply with other QAPD requirements such as QA records.

3.4 Setpoint Control

Instrument and equipment setpoints that could affect nuclear safety shall be controlled in accordance with written instructions. As a minimum, these written instructions shall:

(1) Identify responsibilities and processes for reviewing, approving, and revising setpoints and setpoint changes originally supplied by the reactor plant supplier, the A/E, and the plant's technical staff.

(2) Ensure that setpoints and setpoint changes are consistent with design and accident analysis requirements and assumptions.

(3) Provide for documentation of setpoints, including those determined operationally.

(4) Provide for access to necessary setpoint information for personnel who write or revise plant procedures, operate or maintain plant equipment, develop or revise design documents, or develop or revise accident analyses.

3.5 NQA-1-1994 Commitment/Exceptions

In establishing its program for design control and verification, Dominion commits to compliance with NQA-1-1994, Basic Requirement 3, and Supplement 3S-1, the subsurface investigation requirements in Subpart 2.20, and the standards for computer software in Subpart 2.7.

Section 4 Procurement Document Control

Dominion has established the necessary measures and governing procedures to assure that purchased items and services are subject to appropriate quality and technical requirements. Procurement document changes shall be subject to the same degree of control as utilized in the preparation of the original documents. These controls include provisions such that:

- Where original technical or quality assurance requirements cannot be determined, an engineering evaluation is conducted and documented by qualified staff to establish appropriate requirements and controls to assure that interfaces, interchangeability, safety, fit and function, as applicable, are not adversely affected or contrary to applicable regulatory requirements.
- Applicable technical, regulatory, administrative, quality and reporting requirements (such as specifications, codes, standards, tests, inspections, special processes, and 10 CFR 21) are invoked for procurement of items and services. 10 CFR 21 requirements for posting, evaluating, and reporting will be followed and imposed on suppliers when applicable. Applicable design bases and other requirements necessary to assure adequate quality shall be included or referenced in documents for procurement of items and services. To the extent necessary, procurement documents shall require suppliers to have a documented QA program that is determined to meet the applicable requirements of 10 CFR 50, Appendix B, as appropriate to the circumstances of procurements (or the supplier may work under Dominion's approved QA program).

Reviews of procurement documents shall be performed by personnel who have access to pertinent information and who have an adequate understanding of the requirements and intent of the procurement documents.

4.1 NQA-1-1994 Commitment/Exceptions

In establishing controls for procurement, Dominion commits to compliance with NQA-1-1994, Basic Requirement 4 and Supplement 4S-1, with the following clarifications and exceptions:

- NQA-1-1994, Supplement 4S-1
  - Section 2.3 of this Supplement 4S-1 includes a requirement that procurement documents require suppliers to have a documented QAP that implements NQA-1-1994, Part 1. In lieu of this requirement, Dominion may require suppliers to have a documented supplier QAP that is determined to meet the applicable requirements of 10 CFR 50, Appendix B, as appropriate to the circumstances of the procurement.
  - With regard to service performed by a supplier, Dominion procurement documents may allow the supplier to work under the North Anna Unit 3 QAP, including implementing procedures, in lieu of the supplier having its own QAP.

- Section 3 of this Supplement 4S-1 requires procurement documents to be reviewed prior to bid or award of contract. The quality assurance review of procurement documents is satisfied through review of the applicable procurement specification, including the technical and quality procurement requirements, prior to bid or award of contract. Procurement document changes (e.g., scope, technical or quality requirements) will also receive the quality assurance review.
- Procurement documents for Commercial Grade Items that will be procured by North Anna Unit 3 for use as safety-related items shall contain technical and quality requirements such that the procured item can be appropriately dedicated.

Section 5 Instructions, Procedures, and Drawings

North Anna Unit 3 has established the necessary measures and governing procedures to ensure that activities affecting quality are prescribed by and performed in accordance with instructions, procedures, or drawings of a type appropriate to the circumstances and which, where applicable, include quantitative or qualitative acceptance criteria to implement the QAP as described in the QAPD. Such documents are prepared and controlled according to Part II, Section 6. In addition, means are provided to disseminate to the staff instructions of both general and continuing applicability, as well as those of short-term applicability. Provisions are included for reviewing, updating, and canceling such instructions.

5.1 Procedure Adherence

Dominion's policy is that procedures are followed, and the requirements for use of procedures have been established in administrative procedures. Where procedures cannot be followed as written, provisions are established for making changes in accordance with Part II, Section 6. Requirements are established to identify the manner in which procedures are to be implemented, including identification of those tasks that require: (1) the written procedure to be present and followed step-by-step while the task is being performed, (2) the user to have committed the procedure steps to memory, (3) verification of completion of significant steps, by initials or signatures or use of check-off lists. Procedures that are required to be present and referred to directly are those developed for extensive or complex jobs where reliance on memory cannot be trusted, tasks that are infrequently performed, and tasks where steps must be performed in a specified sequence.

In cases of emergency, personnel are authorized to depart from approved procedures when necessary to prevent injury to personnel or damage to the plant. Such departures are recorded describing the prevailing conditions and reasons for the action taken.

5.2 Procedure Content

The established measures address the applicable content of procedures as described in the introduction to Part II of NQA-1-1994. In addition, procedures governing tests, inspections, operational activities and maintenance will include as applicable, initial conditions and prerequisites for the performance of the activity.

5.3 NQA-1-1994 Commitment

In establishing procedural controls, Dominion commits to compliance with NQA-1-1994, Basic Requirement 5.

Section 6 Document Control

Dominion has established the necessary measures and governing procedures to control the preparation of, issuance of, and changes to documents that specify quality requirements or prescribe how activities affecting quality, including organizational interfaces, are controlled to assure that correct documents are being employed. The control systems (including electronic systems used to make documents available) are documented and provide for the following:

a. identification of documents to be controlled and their specified distribution;

b. a method to identify the correct document (including revision) to be used and control of superseded documents;

c. identification of assignment of responsibility for preparing, reviewing, approving, and issuing documents;

d. review of documents for adequacy, completeness, and correctness prior to approval and issuance;

e. a method for providing feedback from users to continually improve procedures and work instructions; and

f. coordinating and controlling interface documents and procedures.

The types of documents to be controlled include:

a. drawings such as design, construction, installation, and as-built drawings.

b. engineering calculations.

c. design specifications.

d. purchase orders and related documents.

e. vendor-supplied documents.

f. audit, surveillance, and quality verification/inspection procedures.

g. inspection and test reports.

h. instructions and procedures for activities covered by the QAPD including design, construction, installation, operating (including normal and emergency operations), maintenance, calibration, and routine testing.

i. technical specifications.

j. nonconformance reports and corrective action reports.

During the operational phase, where temporary procedures are used, they shall include a designation of the period of time during which it is acceptable to use them.

6.1 Review and Approval of Documents

Documents are reviewed for adequacy by qualified persons other than the preparer. During the construction phase, procedures for design, construction, and installation are also reviewed by the nuclear oversight group to ensure quality assurance measures have been appropriately applied. The documented review signifies concurrence.

During the operations phase, documents affecting the configuration or operation of the station as described in the SAR are screened to identify those that require review by the IRB prior to implementation as described in Part II, Section 2.

To ensure effective and accurate procedures during the operational phase, applicable procedures are reviewed, and updated as necessary, based on the following conditions:

a. following any modification to a system;

b. following an unusual incident, such as an accident, significant operator error, or equipment malfunction;

c. when procedure discrepancies are found;

d. prior to use if not used in the previous two years; or

e. results of QA audits conducted in accordance with Part II, Section 18.1.

Prior to issuance or use, documents including revisions thereto, are approved by the designated authority. A listing of all controlled documents identifying the current approved revision, or date, is maintained so personnel can readily determine the appropriate document for use.

6.2 Changes to Documents

Changes to documents, other than those defined in implementing procedures as minor changes, are reviewed and approved by the same organizations that performed the original review and approval unless other organizations are specifically designated. The reviewing organization has access to pertinent background data or information upon which to base their approval. Where temporary procedure changes are necessary during the operations phase, changes that clearly do not change the intent of the approved procedure may be implemented provided they are approved by two members of the staff knowledgeable in the areas affected by the procedures. Minor changes to documents, such as inconsequential editorial corrections, do not require that the revised documents receive the same review and approval as the original documents. To avoid a possible omission of a required review, the type of minor changes that do not require such a review and approval and the persons who can authorize such a classification are clearly delineated in implementing procedures.

6.3 NQA-1-1994 Commitment

In establishing provisions for document control, Dominion commits to compliance with NQA-1-1994, Basic Requirement 6 and Supplement 6S-1.

Section 7 Control of Purchased Material, Equipment, and Services

Dominion has established the necessary measures and governing procedures to control the procurement of items and services to assure conformance with specified requirements. Such control provides for the following as appropriate: source evaluation and selection, evaluation of objective evidence of quality furnished by the supplier, source inspection, audit, and examination of items or services.

7.1 Acceptance of Item or Service

Dominion establishes and implements measures to assess the quality of purchased items and services, whether purchased directly or through contractors, at intervals and to a depth consistent with the item's or service's importance to safety, complexity, quantity, and the frequency of procurement. Verification actions include testing, as appropriate, during design, fabrication and construction activities. Verifications occur at the appropriate phases of the procurement process, including, as necessary, verification of activities of suppliers below the first tier.

Measures to assure the quality of purchased items and services include the following, as applicable:

- Items are inspected, identified, and stored to protect against damage, deterioration, or misuse.
- Prospective suppliers of safety-related items and services are evaluated to assure that only qualified suppliers are used. Qualified suppliers are audited on a triennial basis. In addition, if a subsequent contract or a contract modification significantly enlarges the scope of, or changes the methods or controls for activities performed by the same supplier, an audit of the modified requirements is conducted, thus starting a new triennial period. North Anna Unit 3 may utilize audits conducted by outside organizations for supplier qualification provided that the scope and adequacy of the audits meet North Anna Unit 3 requirements. Documented annual evaluations are performed for qualified suppliers to assure they continue to provide acceptable products and services. Industry programs, such as those applied by ASME, Nuclear Procurement Issues Committee (NUPIC), or other established utility groups, are used as input or the basis for supplier qualification whenever appropriate. The results of the reviews are promptly considered for effect on a supplier's continued qualification and adjustments made as necessary (including corrective actions, adjustments of supplier audit plans, and input to third party auditing entities, as warranted). In addition, results are reviewed periodically to determine if, as a whole, they constitute a significant condition adverse to quality requiring additional action.

- Provisions are made for accepting purchased items and services, such as source verification, receipt inspection, pre- and post-installation tests, certificates of conformance, and document reviews (including Certified Material Test Report/Certificate). Acceptance actions/documents should be established by the Purchaser with appropriate input from the Supplier and be completed to ensure that procurement, inspection, and test requirements, as applicable, have been satisfied before relying on the item to perform its intended safety function.
- Controls are imposed for the selection, determination of suitability for intended use (critical characteristics), evaluation, receipt and acceptance of commercial-grade services or items to assure they will perform satisfactorily in service in safety-related applications.
- If there is insufficient evidence of implementation of a QA program, the initial evaluation is of the existence of a QA program addressing the scope of services to be provided. The initial audit is performed after the supplier has completed sufficient work to demonstrate that its organization is implementing a QA program.

7.2 NQA-1-1994 Commitment/Exceptions

In establishing procurement verification controls, North Anna Unit 3 commits to compliance with NQA-1-1994, Basic Requirement 7 and Supplement 7S-1, with the following clarifications and exceptions:

- NQA-1-1994, Supplement 7S-1
  - North Anna Unit 3 considers that other 10 CFR 50 licensees, Authorized Nuclear Inspection Agencies, National Institute of Standards and Technology, or other State and Federal agencies which may provide items or services to the Dominion North Anna Unit 3 plant are not required to be evaluated or audited.
  - When purchasing commercial grade calibration services from a calibration laboratory, procurement source evaluation and selection measures need not be performed provided each of the following conditions are met:
    - (1) The purchase documents impose any additional technical and administrative requirements, as necessary, to comply with the North Anna Unit 3 QA program and technical provisions. At a minimum, the purchase document shall require that the calibration certificate/report include identification of the laboratory equipment/standard used.
    - (2) The purchase documents require reporting as-found calibration data when calibrated items are found to be out-of-tolerance.
    - (3) A documented review of the supplier’s accreditation will be performed and will include a verification of each of the following:

- The calibration laboratory holds a domestic (United States) accreditation by any one of the following bodies, which are recognized by the International Laboratory Accreditation Cooperation (ILAC) Mutual Recognition Arrangement (MRA):
  - National Voluntary Laboratory Accreditation Program (NVLAP), administered by the National Institute of Standards & Technology;
  - American Association for Laboratory Accreditation (A2LA);
  - ACLASS Accreditation Services (ACLASS);
  - International Accreditation Service (IAS);
  - Laboratory Accreditation Bureau (L-A-B);
  - Other NRC-approved laboratory accrediting body.
- The accreditation encompasses ANS/ISO/IEC 17025, “General Requirements for the Competence of Testing and Calibration Laboratories.”
- The published scope of accreditation for the calibration laboratory covers the necessary measurement parameters, range, and uncertainties.

•• For Section 8.1, Dominion considers documents that may be stored in approved electronic media under Dominion or vendor control, not physically located on the plant site, but are accessible from the respective nuclear facility site, as meeting the NQA-1 requirement for documents to be available at the site. When construction is complete, sufficient as-built documentation will be turned over to Dominion to support operations. The Dominion records management system will provide for timely retrieval of necessary records.

•• In lieu of the requirements of Section 10, Commercial Grade Items, controls for commercial grade items and services are established in North Anna Unit 3 documents using 10 CFR 21 and the guidance of EPRI NP-5652 as discussed in Generic Letter 89-02 and Generic Letter 91-05.

– For commercial grade items, special quality verification requirements are established and described in Dominion documents to provide the necessary assurance an item will perform satisfactorily in service. The Dominion documents address determining the critical characteristics that ensure an item is suitable for its intended use, technical evaluation of the item, receipt requirements, and quality evaluation of the item.

- Dominion will also use other appropriate approved regulatory means and controls to support Dominion commercial grade dedication activities. One example of this is Electric Power Research Institute (EPRI) Topical Report TR-106439, "Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications," dated July 17, 1997. Dominion will assume 10 CFR 21 reporting responsibility for all items that Dominion dedicates as safety-related.

Section 8 Identification and Control of Materials, Parts, and Components

Dominion has established the necessary measures and governing procedures to identify and control items to prevent the use of incorrect or defective items. This includes controls for consumable materials and items with limited shelf life. The identification of items is maintained throughout fabrication, erection, installation and use so that the item can be traced to its documentation, consistent with the item’s effect on safety. Identification locations and methods are selected so as not to affect the function or quality of the item.

8.1 NQA-1-1994 Commitment

In establishing provisions for identification and control of items, Dominion commits to compliance with NQA-1-1994, Basic Requirement 8 and Supplement 8S-1.

Section 9 Control of Special Processes

Dominion has established the necessary measures and governing procedures to assure that special processes that require interim process controls to assure quality, such as welding, heat treating, and nondestructive examination, are controlled. These provisions include assuring that special processes are accomplished by qualified personnel using qualified procedures and equipment. Personnel are qualified and special processes are performed in accordance with applicable codes, standards, specifications, criteria or other specially established requirements. Special processes are those where the results are highly dependent on the control of the process or the skill of the operator, or both, and for which the specified quality cannot be fully and readily determined by inspection or test of the final product.

9.1 NQA-1-1994 Commitment

In establishing measures for the control of special processes, Dominion commits to compliance with NQA-1-1994, Basic Requirement 9 and Supplement 9S-1.

Section 10 Inspection

Dominion has established the necessary measures and governing procedures to implement inspections that assure items, services, and activities affecting safety meet established requirements and conform to applicable documented specifications, instructions, procedures, and design documents. Inspection may also be applied to items, services, and activities affecting plant reliability and integrity. Types of inspections may include those verifications related to procurement, such as source, in-process, final, and receipt inspection, as well as construction, installation, and operations activities. Inspections are carried out by properly qualified persons independent of those who performed or directly supervised the work. Inspection results are documented.

10.1 Inspection Program

The inspection program establishes inspections (including surveillance of processes), as necessary to verify quality: (1) at the source of supplied items or services, (2) in-process during fabrication at a supplier's facility or at a Company facility, (3) for final acceptance of fabricated and/or installed items during construction, (4) upon receipt of items for a facility, and (5) during maintenance, modification, inservice, and operating activities.

The inspection program establishes requirements for planning inspections, such as the group or discipline responsible for performing the inspection, where inspection hold points are to be applied, determining applicable acceptance criteria, the frequency of inspection to be applied, and identification of special tools needed to perform the inspection. Inspection planning is performed by personnel qualified in the discipline related to the inspection and includes qualified inspectors or engineers. Inspection plans are based on, as a minimum, the importance of the item to the safety of the facility, the complexity of the item, technical requirements to be met, and design specifications. Where significant changes in inspection activities for the facilities are to occur, management responsible for the inspection programs evaluate the resource and planning requirements to ensure effective implementation of the inspection program.

Inspection program documents establish requirements for performing the planned inspections, and documenting required inspection information such as rejection, acceptance, and reinspection results, and the person(s) performing the inspection.

Inspection results are documented by the inspector, reviewed by authorized personnel qualified to evaluate the technical adequacy of the inspection results, and controlled by instructions, procedures, and drawings.

10.2 Inspector Qualification

Dominion has established qualification programs for personnel performing quality inspections. The qualification program requirements are described inPart II, Section 2. These qualification programs are applied to individuals performing quality inspections regardless of the functional group where they are assigned.

10.3 NQA-1-1994 Commitment/Exceptions

In establishing inspection requirements, Dominion commits to compliance with NQA-1-1994, Basic Requirement 10, Supplement 10S-1 and Subpart 2.4, with the following clarification. In addition, Dominion commits to compliance with the requirements of Subparts 2.5 and 2.8 for establishing appropriate inspection requirements.

- Subpart 2.4 commits Dominion to IEEE Std. 336-1985. IEEE Std. 336 1985 refers to IEEE Std. 498-1985. Both IEEE Std. 336-1985 and IEEE Std. 498-1985 use the definition of “Safety Systems Equipment” from IEEE Std. 603-1980. North Anna Unit 3 commits to the definition of Safety Systems Equipment in IEEE Std. 603 1980, but does not commit to the balance of that standard. This definition is only applicable to equipment in the context of Subpart 2.4.
- An additional exception to Subpart 2.4 is addressed in Part II, Section 12 of the QAPD.
- Where inspections at the operating facility are performed by persons within the same organization (e.g., Maintenance group), Dominion takes exception to the requirements of NQA-1-1994, Supplement 10S-2, Section 3.1, in that the inspectors report to the site’s Senior Manager for Safety and Licensing while performing those inspections.

Section 11 Test Control

Dominion has established the necessary measures and governing procedures to demonstrate that items subject to the provisions of the QAPD will perform satisfactorily in service, that the plant can be operated safely and as designed, and that the coordinated operation of the plant as a whole is satisfactory. These programs include criteria for determining when testing is required, such as proof tests before installation, pre-operational tests, post-maintenance tests, post-modification tests, in-service tests, and operational tests (such as surveillance tests required by Plant Technical Specifications), to demonstrate that the performance of plant systems is in accordance with design. Programs also include provisions to establish and adjust test schedules, and to maintain status for periodic or recurring tests. Tests are performed according to applicable procedures that include, consistent with the effect on safety: (1) instructions and prerequisites to perform the test, (2) use of proper test equipment, (3) acceptance criteria, and (4) mandatory verification points as necessary to confirm satisfactory test completion. Test results are documented and evaluated by the organization performing the test and reviewed by a responsible authority to assure that the test requirements have been satisfied. If acceptance criteria are not met, retesting is performed as needed to confirm acceptability following correction of the system or equipment deficiencies that caused the failure.

The initial start-up test program is planned and scheduled to permit safe fuel loading and start-up; to increase power in safe increments; and to perform major testing at specified power levels. If tests require the variation of operating parameters outside of their normal range, the limits within which such variation is permitted will be prescribed. The scope of the testing demonstrates, insofar as practicable, that the plant is capable of withstanding the design transients and accidents. For new facility construction, the suitability of facility operating procedures is checked to the maximum extent possible during the preoperational and initial start-up test programs.

Tests are performed and results documented in accordance with applicable technical and regulatory requirements, including those described in the Technical Specifications and SAR. Test programs ensure appropriate retention of test data in accordance with the records requirements of the QAPD. Personnel that perform or evaluate tests are qualified in accordance with the requirements established in Part II, Section 2.

11.1 NQA-1-1994 Commitment

In establishing provisions for testing, Dominion commits to compliance with NQA-1-1994, Basic Requirement 11 and Supplement 11S-1.

11.2 NQA-1-1994 Commitment for Computer Program Testing

Dominion establishes and implements provisions to assure that computer software used in applications affecting safety is prepared, documented, verified and tested, and used such that the expected output is obtained and configuration control maintained. To this end, Dominion commits to compliance with the requirements of NQA-1-1994, Supplement 11S-2 and Subpart 2.7 to establish the appropriate provisions.

Section 12 Control of Measuring and Test Equipment

Dominion has established the necessary measures and governing procedures to control the calibration, maintenance, and use of measuring and test equipment (M&TE) that provides information important to safe plant operation. The provisions of such procedures cover equipment such as indicating and actuating instruments and gages, tools, reference and transfer standards, and nondestructive examination equipment. The suppliers of commercial-grade calibration services are controlled as described in Part II, Section 7.

12.1 Installed Instrument and Control Devices

For the operations phase of the facilities, Dominion has established and implements procedures for the calibration and adjustment of instrument and control devices installed in the facility. The calibration and adjustment of these devices is accomplished through the facility maintenance programs to ensure the facility is operated within design and technical requirements. Appropriate documentation will be maintained for these devices to indicate the control status, when the next calibration is due, and identify any limitations on use of the device.

12.2 NQA-1-1994 Commitment/Exceptions

In establishing provisions for control of measuring and test equipment, Dominion commits to compliance with NQA-1-1994, Basic Requirement 12 and Supplement 12S-1 with the following clarification and exception:

- The out of calibration conditions described in paragraph 3.2 of Supplement 12S-1 refers to when the M&TE is found out of the required accuracy limits (i.e., out of tolerance) during calibration.
- Measuring and test equipment are not required to be marked with the calibration status where it is impossible or impractical due to equipment size or configuration (such as the label will interfere with operation of the device) provided the required information is maintained in suitable documentation traceable to the device. This exception also applies to the calibration labeling requirement stated in NQA-1-1994, Subpart 2.4, Section 7.2.1 (ANSI/IEEE Std. 336-1985).

Section 13 Handling, Storage, and Shipping

Dominion has established the necessary measures and governing procedures to control the handling, storage, packaging, shipping, cleaning, and preservation of items to prevent inadvertent damage or loss, and to minimize deterioration. These provisions include specific procedures, when required to maintain acceptable quality of the items important to the safe operations of the plant. Items are appropriately marked and labeled during packaging, shipping, handling and storage to identify, maintain, and preserve the item's integrity and indicate the need for special controls. Special controls (such as containers, shock absorbers, accelerometers, inert gas atmospheres, specific moisture content levels and temperature levels) are provided when required to maintain acceptable quality.

Special or additional handling, storage, shipping, cleaning and preservation requirements are identified and implemented as specified in procurement documents and applicable procedures. Where special requirements are specified, the items and containers (where used) are suitably marked.

Special handling tools and equipment are used and controlled as necessary to ensure safe and adequate handling. Special handling tools and equipment are inspected and tested at specified time intervals and in accordance with procedures to verify that the tools and equipment are adequately maintained.

Operators of special handling and lifting equipment are experienced or trained in the use of the equipment. During the operational phase, Dominion establishes and implements controls over hoisting, rigging and transport activities to the extent necessary to protect the integrity of the items involved, as well as potentially affected nearby structures and components. Where required, Dominion complies with applicable hoisting, rigging and transportation regulations and codes.

13.1 Housekeeping

Housekeeping practices are established to account for conditions or environments that could affect the quality of structures, systems and components within the plant. This includes control of cleanness of facilities and materials, fire prevention and protection, disposal of combustible material and debris, control of access to work areas, protection of equipment, radioactive contamination control and storage of solid radioactive waste. Housekeeping practices help assure that only proper materials, equipment, processes and procedures are used and that the quality of items is not degraded. Necessary procedures or work instructions, such as for electrical bus and control center cleaning, cleaning of control consoles, and radioactive decontamination are developed and used.

13.2 NQA-1-1994 Commitment/Exceptions

In establishing provisions for handling, storage and shipping, Dominion commits to compliance with NQA-1-1994, Basic Requirement 13 and Supplement 13S-1. Dominion also commits, during the construction and pre-operational phase of the plant, to compliance with the requirements of NQA-1-1994, Subpart 2.1, Subpart 2.2, and Subpart 3.2, Appendix 2.1, with the following clarifications and exceptions:

- NQA -1-1994, Subpart 2.2
  - Subpart 2.2, Section 6.6, “Storage Records:” This section requires written records be prepared containing information on personnel access. As an alternative to this requirement, North Anna Unit 3 documents establish controls for storage areas that describe those authorized to access areas and the requirements for recording access of personnel. However, these records of access are not considered quality records and will be retained in accordance with the administrative controls of the applicable plant.
  - Subpart 2.2, Section 7.1 refers to Subpart 2.15 for requirements related to handling of items. The scope of Subpart 2.15 includes hoisting, rigging and transporting of items for the nuclear power plant during construction.
- NQA-1-1994, Subpart 3.2
  - Subpart 3.2, Appendix 2.1: Only Section 3 precautions are being committed to in accordance with RG 1.37. In addition, a suitable chloride stress-cracking inhibitor should be added to the fresh water used to flush systems containing austenitic stainless steels.

Section 14 Inspection, Test, and Operating Status

Dominion has established the necessary measures and governing procedures to identify the inspection, test, and operating status of items and components subject to the provisions of the QAPD in order to maintain personnel and reactor safety and avoid inadvertent operation of equipment. Where necessary to preclude inadvertent bypassing of inspections or tests, or to preclude inadvertent operation, these measures require the inspection, test, or operating status be verified before release, fabrication, receipt, installation, test, or use. These measures also establish the necessary authorities and controls for the application and removal of status indicators or labels.

In addition, temporary design changes (temporary modifications), such as temporary bypass lines, electrical jumpers and lifted wires, and temporary trip-point settings, are controlled by procedures that include requirements for appropriate installation and removal, independent/concurrent verifications, and status tracking.

Administrative procedures also describe the measures taken to control altering the sequence of required tests, inspections, and other operations. Review and approval for these actions is subject to the same control as taken during the original review and approval of tests, inspections, and other operations.

14.1 NQA-1-1994 Commitment

In establishing measures for control of inspection, test, and operating status, Dominion commits to compliance with NQA-1-1994, Basic Requirement 14.

Section 15 Nonconforming Materials, Parts, or Components

Dominion has established the necessary measures and governing procedures to control items, including services, that do not conform to specified requirements to prevent inadvertent installation or use. Controls provide for identification, documentation, evaluation, segregation when practical, and disposition of nonconforming items, and for notification to affected organizations. Controls are provided to address conditional release of nonconforming items for use on an at-risk basis prior to resolution and disposition of the nonconformance, including maintaining identification of the item and documenting the basis for such release. Conditional release of nonconforming items for installation requires the approval of the designated management. Nonconformances are corrected or resolved prior to depending on the item to perform its intended safety function. Nonconformances are evaluated for impact on operability of quality structures, systems, and components to assure that the final condition does not adversely affect safety, operation, or maintenance of the item or service. Nonconformances to design requirements dispositioned repair or use-as-is are subject to design control measures commensurate with those applied to the original design. Nonconformance dispositions are reviewed for adequacy, analysis of quality trends, and reports provided to the designated management. Significant trends are reported to management in accordance with Dominion procedures, regulatory requirements, and industry standards.

15.1 Reporting Program

Dominion has the necessary measures and governing procedures that implement a reporting program that conforms to the requirements of 10 CFR 52, 10 CFR 50.55 and/or 10 CFR 21 during COL design and construction, and 10 CFR 21 during operations.

15.2 NQA-1-1994 Commitment

In establishing measures for nonconforming materials, parts, or components, Dominion commits to compliance with NQA-1-1994, Basic Requirement 15, and Supplement 15S-1.

Section 16 Corrective Action

Dominion has established the necessary measures and governing procedures to promptly identify, control, document, classify, and correct conditions adverse to quality. Dominion procedures assure that corrective actions are documented and initiated following the determination of conditions adverse to quality in accordance with regulatory requirements and applicable quality standards. Dominion procedures require personnel to identify known conditions adverse to quality. When complex issues arise where it cannot be readily determined if a condition adverse to quality exists, Dominion documents establish the requirements for documentation and timely evaluation of the issue. Reports of conditions adverse to quality are analyzed to identify trends. Significant conditions adverse to quality and significant adverse trends are documented and reported to responsible management. In the case of a significant condition adverse to quality, the cause is determined and actions to preclude recurrence are taken.

In the case of suppliers working on safety-related activities, or other similar situations, Dominion may delegate specific responsibilities of the Corrective Action program but Dominion maintains responsibility for the program's effectiveness.

16.1 Reporting Program

Dominion has the necessary measures and governing procedures that implement a reporting program that conforms to the requirements of 10 CFR 52, 10 CFR 50.55 and/or 10 CFR Part 21, during COL design and construction, and 10 CFR 21 during operations.

16.2 NQA-1-1994 Commitment

In establishing provisions for corrective action, Dominion commits to compliance with NQA-1-1994, Basic Requirement 16.

Section 17 Quality Assurance Records

Dominion has the necessary measures and governing procedures to ensure that sufficient records of items and activities affecting quality are developed, reviewed, approved, issued, used, and revised to reflect completed work. The provisions of such procedures establish the scope of the records retention program for Dominion and include requirements for records administration, including receipt, preservation, retention, storage, safekeeping, retrieval, access controls, user privileges, and final disposition.

17.1 Record Retention

Measures are established that ensure that sufficient records of completed items and activities affecting quality are appropriately stored. Such records and their retention times are defined in appropriate procedures. In all cases where state, local, or other agencies have more restrictive requirements for record retention, those requirements will be met.

17.2 Electronic Records

When using electronic records storage and retrieval systems, Dominion complies with NRC guidance Generic Letter 88-18, “Plant Record Storage on Optical Disks.” Dominion will manage the storage of QA Records in electronic media consistent with the intent of RIS 2000-18 and associated NIRMA Guidelines TG 11-1998, TG15-1998, TG16-1998, and TG21-1998.

17.3 NQA-1-1994 Commitment/Exceptions

In establishing provisions for records, Dominion commits to compliance with NQA-1-1994, Basic Requirement 17 and Supplement 17S-1, with the following clarifications and exceptions:

- NQA-1-1994, Supplement 17S-1
  - Supplement 17S-1, Section 4.2(b) requires records to be firmly attached in binders or placed in folders or envelopes for storage in steel file cabinets or on shelving in containers. For hard-copy records maintained by Dominion, the records are suitably stored in steel file cabinets or on shelving in containers, except that methods other than binders, folders, or envelopes may be used to organize the records for storage.

Section 18 Audits

Dominion has established the necessary measures and governing procedures to implement audits to verify that activities covered by the QAPD are performed in conformance with the requirements established. The audit programs are themselves reviewed for effectiveness as a part of the overall audit process.

18.1 Performance of Audits

Internal audits of selected aspects of licensing, design, construction phase and operating activities are performed with a frequency commensurate with safety significance and in a manner which assures that audits of safety-related activities are completed. During the early portions of North Anna Unit 3 COL activities, audits will focus on areas including, but not limited to, site investigation, procurement, and corrective action. Functional areas of an organization's QA program for auditing include, at a minimum, verification of compliance and effectiveness of implementation of internal rules, procedures (e.g., operating, design, procurement, maintenance, modification, refueling, surveillance, and test), Technical Specifications, regulations and license conditions, programs for training, retraining, qualification and performance of operating staff, corrective actions, and observation of performance of operating, refueling, maintenance and modification activities, including associated recordkeeping.

The audits are scheduled on a formal preplanned audit schedule. The audit system is reviewed periodically and revised as necessary to assure coverage commensurate with current and planned activities. Additional audits may be performed as deemed necessary by management. The scope of the audit is determined by the quality status and safety importance of the activities being performed. These audits are conducted by trained personnel not having direct responsibilities in the area being audited and in accordance with preplanned and approved audit plans or checklists, under the direction of a qualified lead auditor and the cognizance of the manager for the North Anna Unit 3 nuclear oversight group.

Dominion is responsible for conducting periodic internal and external audits. Internal audits are conducted to determine the adequacy of programs and procedures (by representative sampling), and to determine if they are meaningful and comply with the overall QAPD. External audits determine the adequacy of supplier and contractor quality assurance program.

The results of each audit are reported in writing to the CNO, and the executives responsible for the area audited. Additional internal distribution is made to other concerned management levels in accordance with approved procedures.

Management responds to all audit findings and initiates corrective action where indicated. Where corrective action measures are indicated, documented follow-up of applicable areas through inspections, review, re-audits, or other appropriate means is conducted to verify implementation of assigned corrective action.

Audits of suppliers of safety-related components and/or services are conducted as described in Section 7.1.

18.2 Internal Audits

Internal audits of organization and facility activities, conducted prior to placing the facility in operation, should be performed in such a manner as to assure that an audit of all applicable QA program elements is completed for each functional area at least once each year or at least once during the life of the activity, whichever is shorter.

Internal audits of activities, conducted after placing the facility in operation, should be performed in such a manner as to assure that an audit of all applicable QA program elements is completed for each functional area within a period of two years. Internal audit frequencies of well established activities, conducted after placing the facility in operation, may be extended one year at a time beyond the above two-year interval based on the results of an annual evaluation of the applicable functional area and objective evidence that the functional area activities are being satisfactorily accomplished. The evaluation should include a detailed performance analysis of the functional area based upon applicable internal and external source data and due consideration of the impact of any functional area changes in responsibility, resources. or management. However, the internal audit frequency interval should not exceed a maximum of four years. If an adverse trend is identified in the applicable functional area, the extension of the internal audit frequency interval should be rescinded and an audit scheduled as soon as practicable.

During the operations phase, audits are performed at a frequency commensurate with the safety significance of the activities and in such a manner to assure audits of all applicable QA program elements are completed within a period of two years. These audits will include, as a minimum, activities in the following areas:

(1) The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions including administrative controls.

(2) The performance, training, and qualifications of the facility staff.

(3) The performance of activities required by the QAPD to meet the criteria of 10 CFR 50, Appendix B.

(4) The Fire Protection Program and implementing procedures. A fire protection equipment and program implementation inspection and audit are conducted utilizing either a qualified offsite licensed fire protection engineer or an outside qualified fire protection consultant.

(5) Other activities and documents considered appropriate by the corporate executive for nuclear operations, or the CNO.

Audits may also be used to meet the periodic review requirements of the code for the Security, Emergency Preparedness, and Radiological Protection programs within the provisions of the applicable code.

Internal audits include verification of compliance and effectiveness of the administrative controls established for implementing the requirements of the QAPD; regulations and license provisions; provisions for training, retraining, qualification, and performance of personnel performing activities covered by the QAPD; corrective actions taken following abnormal occurrences; and, observation of the performance of construction, fabrication, operating, refueling, maintenance and modification activities including associated record keeping.

18.3 NQA-1-1994 Commitment

In establishing the independent audit program, Dominion commits to compliance with NQA-1-1994, Basic Requirement 18 and Supplement 18S-1.

Part III Nonsafety-Related SSC Quality Control

Section 1 Nonsafety-Related SSCs - Significant Contributors to Plant Safety

Specific program controls are applied to nonsafety-related SSCs, for which 10 CFR 50, Appendix B is not applicable, that are significant contributors to plant safety. The specific program controls consistent with applicable sections of the QAPD are applied to those items in a selected manner, targeted at those characteristics or critical attributes that render the SSC a significant contributor to plant safety.

The following clarify the applicability of the QA Program to the nonsafety-related SSCs and related activities, including the identification of exceptions to the QA Program described in Part II, Sections 1 through 18 taken for nonsafety-related SSCs.

1.1 Organization

Verification activities described in this part may be performed by the Dominion line organization, the QA organization described in Part II is not required to perform these functions.

1.2 QA Program

Dominion QA requirements for nonsafety-related SSCs are established in the QAPD and appropriate procedures. Suppliers of these SSCs or related services describe the quality controls applied in appropriate procedures. A new or separate QA program is not required.

1.3 Design Control

Dominion has design control measures to ensure that the contractually established design requirements are included in the design. These measures ensure that applicable design inputs are included or correctly translated into the design documents, and deviations from those requirements are controlled. Design verification is provided through the normal supervisory review of the designer’s work.

1.4 Procurement Document Control

Procurement documents for items and services obtained by or for Dominion include or reference documents describing applicable design bases, design requirements, and other requirements necessary to ensure component performance. The procurement documents are controlled to address deviations from the specified requirements.

1.5 Instructions, Procedures, and Drawings

Dominions provides documents such as, but not limited to, written instructions, plant procedures, drawings, vendor technical manuals, and special instructions in work orders, to direct the performance of activities affecting quality. The method of instruction employed provides an appropriate degree of guidance to the personnel performing the activity to achieve acceptable functional performance of the SSC.

1.6 Document Control

Dominion controls the issuance and change of documents that specify quality requirements or prescribe activities affecting quality to ensure that correct documents are used. These controls include review and approval of documents, identification of the appropriate revision for use, and measures to preclude the use of superseded or obsolete documents.

1.7 Control of Purchased Items and Services

Dominion employs measures, such as inspection of items or documents upon receipt or acceptance testing, to ensure that all purchased items and services conform to appropriate procurement documents.

1.8 Identification and Control of Purchased Items

Dominion employs measures where necessary, to identify purchased items and preserve their functional performance capability. Storage controls take into account appropriate environmental, maintenance, or shelf life restrictions for the items.

1.9 Control of Special Processes

Dominion employs process and procedure controls for special processes, including welding, heat treating, and nondestructive testing. These controls are based on applicable codes, standards, specifications, criteria, or other special requirements for the special process.

1.10 Inspection

Dominion uses documented instructions to ensure necessary inspections are performed to verify conformance of an item or activity to specified requirements or to verify that activities are satisfactorily accomplished. These inspections are performed by knowledgeable personnel who may be in the same line organization as those performing the activity being inspected.

1.11 Test Control

Dominion employs measures to identify required testing that demonstrates that equipment conforms to design requirements. These tests are performed in accordance with test instructions or procedures. The test results are recorded, and authorized individuals evaluate the results to ensure that test requirements are met.

1.12 Control of Measuring and Test Equipment (M&TE)

Dominion employs measures to control M&TE use, and calibration and adjustment at specific intervals or prior to use.

1.13 Handling, Storage, and Shipping

Dominion employs measures to control the handling, storage, cleaning, packaging, shipping, and preservation of items to prevent damage or loss, and to minimize deterioration. These measures include appropriate marking or labels, and identification of any special storage or handling requirements.

1.14 Inspection, Test, and Operating Status

Dominion employs measures to identify items that have satisfactorily passed required tests and inspections and to indicate the status of inspection, test, and operability as appropriate.

1.15 Control of Nonconforming Items

Dominion employs measures to identify and control items that do not conform to specified requirements to prevent their inadvertent installation or use.

1.16 Corrective Action

Dominion employs measures to ensure that failures, malfunctions, deficiencies, deviations, defective components, and nonconformances are properly identified, reported, and corrected.

1.17 Records

Dominion employs measures to ensure records are prepared and maintained to furnish evidence that the above requirements for design, procurement, document control, inspection, and test activities have been met.

1.18 Audits

Dominion employs measures for line management to periodically review and document the adequacy of the process, including taking any necessary corrective action. Audits independent of line management are not required. Line management is responsible for determining whether reviews conducted by line management or audits conducted by any organization independent of line management are appropriate. If performed, audits are conducted and documented to verify compliance with design and procurement documents, instructions, procedures, drawings, and inspection and test activities. Where the measures of this part (Part III) are implemented by the same programs, processes, or procedures as the comparable activities of Part II, the audits performed under the provisions of Part II may be used to satisfy the review requirements of this Section (Part III, Section 1.18).

Section 2 Nonsafety-Related SSCs Credited for Regulatory Events

The following criteria apply to fire protection (10 CFR 50.48), anticipated transients without scram (ATWS) (10 CFR 50.62), and the station blackout (SBO) (10 CFR 50.63) SSCs that are not safety-related:

Dominion implements quality requirements for the fire protection system in accordance with Regulatory Position 1.7, “Quality Assurance,” in Regulatory Guide 1.189, “Fire Protection for Operating Nuclear Power Plants.”

Dominion implements the quality requirements for ATWS equipment in accordance with Generic Letter 85-06, “Quality Assurance Guidance for ATWS Equipment That Is Not Safety Related.”

Dominion implements quality requirements for SBO equipment in accordance with Regulatory Position 3.5, “Quality Assurance and Specific Guidance for SBO Equipment That Is Not Safety Related,” and Appendix A, “Quality Assurance Guidance for Non-Safety Systems and Equipment,” in Regulatory Guide 1.155, “Station Blackout.”

Part IV Regulatory Commitments

Section 1 NRC Regulatory Guides and Quality Assurance Standards

This section identifies the NRC Regulatory Guides and the other quality assurance standards which have been selected to supplement and support the North Anna Unit 3 QAPD. North Anna Unit 3 commits to compliance with these standards to the extent described herein. Commitment to a particular Regulatory Guide or other QA standard does not constitute a commitment to the Regulatory Guides or QA standards that may be referenced therein.

1.1 Regulatory Guides

Regulatory Guide 1.26, Revision 4, March 2007- Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

Regulatory Guide 1.26 defines classification of systems and components.

Dominion commits to the applicable regulatory position guidance provided in this regulatory guide for North Anna Unit 3 components outside the scope of the DCD. The requirements for quality group classifications and standerds defined by the DCD meet the regulatory guidance of Revision 3.

Regulatory Guide 1.26, Revision 3, February 1976 - Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

Regulatory Guide 1.26 defines classification of systems and components.

Dominion commits to the applicable regulatory position guidance provided in this regulatory guide for North Anna Unit 3 components within the scope of the DCD with the exceptions described in the ESBWR DCD Table 1.9-21, Table 1.9-21a, and Table 1.9-21b.

Regulatory Guide 1.29, Revision 4, March 2007- Seismic Design Classification

Regulatory Guide 1.29 defines systems required to withstand a safe shutdown earthquake (SSE).

Dominion commits to the applicable regulatory position guidance provided in this regulatory guide for North Anna Unit 3 systems outside the scope of the DCD. The requirements for seismic design classification defined by the DCD meet the regulatory guidance of Revision 3.

Regulatory Guide 1.29, Revision 3, September 1978 - Seismic Design Classification

Regulatory Guide 1.29 defines systems required to withstand a safe shutdown earthquake (SSE).

Dominion commits to the applicable regulatory position guidance provided in this regulatory guide for North Anna Unit 3 systems within the scope of the DCD with the exceptions described in the ESBWR DCD Table 1.9-21, Table 1.9-21a, and Table 1.9-21b.

Regulatory Guide 1.37, Revision 1, March 2007 - Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

Regulatory Guide 1.37 provides guidance on specifying water quality and precautions related to the use of alkaline cleaning solutions and chelating agents.

Dominion commits to the applicable regulatory position guidance provided in this regulatory guide for North Anna Unit 3 during the construction and preoperational phase of the plant.

1.2 Standards

ASME NQA-1-1994 Edition - Quality Assurance Requirements for Nuclear Facility Applications

Dominion commits to NQA-1-1994, Parts I and II, as described in the foregoing sections of this document.

Nuclear Information and Records Management Association, Inc. (NIRMA) Technical Guides (TGs)

Dominion commits to NIRMA TGs as described in Part II, Section 17.

Chapter 18 Human Factors Engineering

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

18.13 Human Performance Monitoring

18.13.3 Elements of Human Performance Monitoring Process

Delete the first sentence in the fourth paragraph. Add the following to the end of this section:

STD COL 18.13-1-H The HPM program will be implemented prior to the beginning of the first licensed operator training class.

18.13.5 COL Information

18.13-1-H Milestone for HPM Implementation

STD COL 18.13-1-H This COL item is addressed in Section 18.13.3.

Chapter 19 Probabilistic Risk Assessment and Severe Accidents

19.1 Introduction

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

19.2 PRA Results and Insights

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

19.2.3.2.4 Evaluation of External Event Seismic

Significant Core Damage Sequences of External Event Seismic

Replace the second and third sentences of the first paragraph with the following.

STD COL 19.2.6-1-H As-built SSC High Confidence Low Probability of Failure (HCLPF)s will be compared to those assumed in the ESBWR seismic margin analysis shown in DCD Table 19.2-4. Deviations from the HCLPF values or other assumptions in the seismic margins evaluation will be analyzed to determine if any new vulnerabilities have been introduced. This comparison and analysis will be completed prior to fuel load.

19.2.6 COL Information

19.2.6-1-H Seismic High Confidence Low Probability of Failure Margins

STD COL 19.2.6-1-H This COL Item is addressed in Section 19.2.3.2.4.

19.3 Severe Accident Evaluations

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

19.4 PRA Maintenance

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

19.5 Conclusions

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

NAPS SUP 19.5-1

In accordance with 10 CFR 52.79(a)(46), this report is required to contain a description of the plant-specific PRA and its results. As part of the development of the certified design PRA, site and plant-specific information were reviewed to determine if any changes from the certified design PRA were warranted. This review included consideration of site-specific information such as site meteorological data and site-specific population distributions, as well as plant-specific design information that replaced conceptual design information described in the DCD. Section 1.8.5 was also reviewed to determine if there were any departures affecting the PRA results. This review is summarized in Appendix 19AA.

The review of site-specific information and plant-specific design information determined that: 1) the DCD PRA bounds site-specific and plant-specific design parameters and design features and 2) these parameters and features have no significant impact on the DCD PRA results and insights. Therefore, based on this review, it is concluded that there is no significant change from the certified design PRA. In that there are no significant changes from the certified design PRA, incorporation of DCD Chapter 19 into the FSAR satisfies the requirement of 10 CFR 52.79(a)(46) for a description of the plant-specific PRA and its results.

Appendix 19A Regulatory Treatment of Non-Safety Systems (RTNSS)

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

Appendix 19ACM Availability Controls Manual

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

Appendix 19B Deterministic Analysis for Containment Pressure Capability

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

Appendix 19C Probabilistic Analysis for Containment Pressure Fragility

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

NAPS SUP 19.5-1

Appendix 19AA Summary of Plant-Specific PRA Review

19AA.1 Introduction

In accordance with 10 CFR 52.79(a)(46), this appendix provides a summary of the plant-specific PRA and its results.

19AA.2 Development of the ESBWR and Plant-Specific PRAs

The ESBWR PRA used the following North Anna site-specific PRA information to develop bounding PRA parameters:

- Loss of Preferred Power (LOPP) frequency - to determine if the site has unusual off-site power availability problems. The LOPP frequency is divided into plant-centered, switchyard, grid-related, and weather-related initiating events.
- Loss of Service Water frequency - to determine if any unusual characteristics would apply to a particular site, with consideration to loss of ultimate heat sink, and the effects of extreme seasonal temperatures.
- Seismic fragilities - to determine if Early Site Permit fragilities can be applied. Note that High Confidence Low Probability of Failure (HCLPF) values will be confirmed as described in Section 19.2.3.2.4.
- Other Known Site-Specific Issues - to identify site-specific initiating events that are not identified in the ESBWR PRA, such as unique offsite consequence issues.

These parameters represent site-specific features that have the potential to affect the PRA. To ensure that the ESBWR PRA is a bounding standard design, the site-specific values for these parameters were used to develop the ESBWR PRA standard values.

The ESBWR LOPP frequencies are based on NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants Analysis of Loss of Offsite Power Events: 1986-2004." The Grand Gulf and North Anna LOPP frequencies were compared to the ESBWR frequencies to identify any outliers. The data shows that grid-related losses of power are significantly more frequent than plant-centered, switchyard, or weather-related losses of power. Although there is a variance in the values for the LOPP frequencies, their range is acceptable. The conclusions in ESBWR DCD Section 19.2.3.1, Risk from Internal Events, remain valid for the minor variances in LOPP frequencies.

The ESBWR Loss of Service Water frequency is based on NUREG/CR-5750, "Rates of Initiating Events at U. S. Nuclear Power Plants: 1987-1995." The contribution of Loss of Service Water is less than one percent of core damage frequency (CDF). Variances between the reported values depend on the design configuration (e.g., redundancy) of the current plants versus the ESBWR design, or external influences such as loss or degradation of heat sink. Although there is a variance in the values for the Loss of Service Water frequencies, their range is acceptable. The conclusions in DCD Section 19.2.3.1, Risk from Internal Events, also remain valid for the minor variances in Loss of Service Water frequencies.

The ESBWR design incorporates a seismic response spectrum that bounds the potential U.S. sites. The conclusions in DCD Section 19.2.3.2.4, Evaluation of External Event Seismic, remain valid for site-specific differences in seismic response.

There are no unusual terrain features that would affect meteorological data or plume dispersion. The conclusions in DCD Section 19.2.5 for offsite consequences remain valid for any potential differences between site features.

In addition to the bounding treatment of PRA parameters, there are no departures from the standard design in any systems considered in the PRA model. Therefore, there are no site-specific design features that affect the PRA because the boundary of the certified design covers all of the SSCs necessary for the PRA.

19AA.3 Internal Flooding

19AA.3.1 Internal Flooding Associated with the Yard Area

The yard flood zone is essentially all outside areas of the site, and thus the site plot drawing (FSAR Figure 2.1-201) illustrates the areas of concern. In addition DCD Section 3.4.1.1 stipulates that the plant grade level is above the design flood level. The only components located in the yard that support a safety function are the manual fire hose connections to the Reactor Building and Fuel Building. They provide the capability to connect another source of water to the IC/PCCS pools and the Spent Fuel Pool after seven days following a postulated accident. This timeframe is beyond the time required to be considered for the PRA; therefore, external flooding in the yard does not affect PRA equipment.

19AA.3.2 Internal Flooding Associated with the Service Water Building

The Service Water Structure is a site-specific design feature. It is treated in a bounding manner in the ESBWR PRA to demonstrate that site-specific differences in Service Water Structure design do not have a significant effect on the PRA results. The Service Water Structure houses the four Service Water pumps and their associated power supplies and controls. Because Service Water is a RTNSS function, in accordance with DCD Table 19A-4, the design and installation of the Service Water Structure is required to include protection from the effects of external and internal flooding.

In the ESBWR PRA model, the Service Water Structure is conservatively considered to be one flood zone. All four pumps are assumed to fail in an internal flood. Thus, the ESBWR PRA is bounding for design differences in the Service Water Structure. In addition, the ESBWR PRA model does not credit operator actions to mitigate a flooding event, so differences in building location are not significant.

The conclusion in DCD Section 19.2.3.2.2 is that there are no significant flood-initiated accident sequences due to the low CDF. Overall, the potential effects of Service Water Structure design differences are accounted for by using a bounding analysis, and therefore, are not significant to the ESBWR PRA.

In summary, the ESBWR PRA provides a reasonable representation of the parameters and conditions that are specific to the North Anna site.