UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 – 0001

April 15, 2011

MEMORANDUM TO: ACRS Members

FROM: Weidong Wang, Senior Staff Engineer ***/RA/***
Reactor Safety Branch B, ACRS

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE ON THE AP1000 REACTOR, JANUARY 10-11, 2011, ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on April 6, 2011, as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As Stated

Cc w/o Attachment: E. Hackett
Y, Dias-Sanabria

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 – 0001

April 6, 2011

MEMORANDUM TO: Weidong Wang, ACRS staff

FROM: Harold B. Ray, Chairman
ACRS AP1000 Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE ON THE AP1000 REACTOR, JANUARY 10-11, 2011, ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting held on January 10-11, 2010, are an accurate record of the proceedings.

/RA/ *04/06/2011*
Harold B. Ray, Chairman *Date*
ACRS AP1000 Subcommittee

Issued: April 6, 2011 Certified: April 6, 2011
Certified by Harold Ray

REVISION 17 TO AP1000 DESIGN CONTROL DOCUMENT
And
VOGTLE ELECTRIC GENERATING PLANT COMBINED OPERATING LICENSE APPLICATIONS

January 10-11, 2011
ROCKVILLE, MARYLAND

INTRODUCTION

The Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on the Westinghouse Electrical Company's AP1000 advanced pressurized water reactor (PWR) design met in Room T-2B1 at the Headquarters of the U.S. Nuclear Regulatory Commission (NRC), located at 11545 Rockville Pike, Rockville, Maryland, on January 10-11, 2011. The purposes of this meeting were to review chapters of the Virgil C. Summer (Summer) Subsequent COL (SCOL) application and its associated advanced Final Safety Evaluation Report (FSER). The Subcommittee was briefed by and held discussions with representatives of South Carolina Electric & Gas (SCE&G) and the U.S. Nuclear Regulatory Commission (NRC) on this subject. As part of the respective review processes, NRC's regulations under 10 CFR Part 52 direct the staff to consult with the ACRS on safety issues before any reactor design can be certified or any NRC operating license can be approved.

The staff's SER review was organized based on the various chapters found in NUREG- 0800 – NRC's "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition." To this end, the Subcommittee planned to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee of the ACRS at a later date. This was the second Subcommittee meeting on the Summer SCOL.

The Chairman for this ACRS Subcommittee was Mr. Harold Ray. Mr. Weidong Wang was the cognizant ACRS staff engineer for this topic and served as the Designated Federal Official for this meeting. Peter Wen, an ACRS staff engineer, supported this two-day meeting as well. The meeting was open to public attendance for most of time except one topic on Loss of Large Area due to Fire or Explosion.

ATTENDEES

ACRS

H. Ray, Subcommittee Chairman	S. Banerjee, Member	D. Bley, Member
C. Brown, Member	Joy Rempe, Member	S. Armijo, Member
M. Ryan, Member	B. Hinze, Invited ACRS Consultant	W. Wang, ACRS Staff

P. Wen, ACRS Staff

The NRC Staff

J. Sebrosky	D. Habib	K. See
G. Stirewalt	D. McGovern	R. Raione
D. Barss	J. Martin	G. Galletta
A. Bowers	C. Cook	P. Pieringer
N. Wright	G. Cicotte	E. Roach
A. Minarik	E. Robinson	D. Brown
D. Sisk	N. Chokshi	R. Patel
L. Wheeler	T. Dinh	P. Hernandez
M. Hayes	J. Budzynski	J. Budzynski
V. Thomas	M. Patterson	D. Terao
Y. Li	E. Sastre	

Others

A. Monroe, SCE&G	R. Whorton, SCE&G	S. Summer, SCE&G
T. Schmidt, SCE&G	E. Bonnette, SCE&G	N. Haggerty, NuStart
W. Hutchins, WEC	D. Patton, Bechtel (SCE&G)	M. Richmond, Bechel (SCE&G)
A. Findikokis	K. Marsh, SCE&G	S. Byrne, SCE&G
B. Williamson, SCE&G	M. Stella, WEC	T. Ray, WEC
E. Cummins, WEC	R. Ziesing, WEC	M. Fayer, PNNL
J. DeBlasio, WEC	T. Pilo, Progress Energy	M. Melton, WEC

The other Individuals and their affiliations attending this meeting are listed in the sign-in sheets in Attachment 1.

SCHEDULED PRESENTATIONS

The detailed agenda identifying the specific presentation topics can be found in Attachment 2. Both during and following the scheduled presentations, the speakers responded to specific

questions and comments from the ACRS Subcommittee members. The scope of the questions, comments, and the speaker's responses had been captured in the verbatim meeting transcripts.

ACRS Subcommittee meeting transcripts can be found at the following NRC Internet website location: http://www.nrc.gov/reading-rm/doc-collections/acrs/tr/subcommittee/.

Opening Remarks

AP1000 Subcommittee Chairman Ray made the opening remarks. He stated that the Subcommittee will review Virgil C. Summer's SCOL application. The NRO staff and the applicant presented the Summer FSAR Chapter 2 (except Section 2.4) evaluation at the AP1000 Subcommittee meeting in July 2010. The staff and applicant will present the rest of the application in this January 2011 meeting. For the agenda item on Loss of a Large Area due to Fire Explosion, the presentation will be closed to the public in order to discuss information that is proprietary to the applicant and its contractors or that is security-related information. Chairman Ray also stated that the full Committee will review an application for the Vogtle reference COL later this week, and he asked Members if they had any items that they would like to direct to those representing Vogtle who were there.

Mr. Frank Akstulewicz, the Deputy Director for Licensing Operations in the Division of New Reactor Licensing, commented that it would begin to see the benefits of standardization as the NRC moves through this application review in a way that the NRC hoped to reduce the demands for time on the Committee Members and the staff.

Mr. Steve Byrne, the Executive Vice President for Generation for SCE&G, made the opening comments as well. He introduced himself as the company's chief nuclear officer since 2005 when they embarked on this new nuclear build path. SCE&G had made excellent progress at the site in preparing for the onset of nuclear safety-related construction activities and looked forward to the successful completion of this licensing process and the start of nuclear safety-related construction activities.

Followed with the opening comments, the applicant and the staff made presentations. The briefing slides with non-proprietary information can be found in Attachment 3.

Key Discussion Points

Member Brown asked how the DCD Rev. 18 would be brought into the SCOL application since the current COL application referenced DCD Rev. 17. SCE&G responded that it would be making another COL submittal with revision to reference DCD Rev. 18. The staff further clarified the processes for the DCD Rev. 18 confirmatory items and any other future DCD changes. If there were significant changes or departures, the staff would bring them back to the ACRS for review. The staff took an action and further clarified this process with a block process diagram in the next day.

For Section 2.4, Hydrologic Engineering, the applicant presented six COL information items, which included hydrological description, floods, cooling water supply, groundwater, accidental release of liquid effluents into ground and surface water, and flood protection emergency operation procedures. The staff presented its evaluations on this section. It reviewed various flooding mechanisms to determine the site characteristics related to the design-basis flood and the required flooding protection. The staff postulated a dam failure scenario in addition to those discussed in the

FSAR by the applicant and determined that peak elevations from the postulated breach would not exceed the site grade elevation. The staff also stated that it reviewed and determined acceptable for the evaluation by the applicant of the postulated accident release from the radwaste management system and its potential effects on groundwater and surface water.

For Section 13.3, Emergency Planning, the applicant discussed DCD departures, emergency plan design, site layout, command-and-control emergency facilities, emergency response, emergency planning zone, and offsite education and alerting. The DCD departures were the location selections for the Technical Support Center (TSC) and Operational Support Center (OSC). The TSC would be located in the New Nuclear Operations Building, which would be located by Unit 1 and constructed in March of this year. It would house the Unit 1 TSC. As Units 2 and 3 come online, that TSC would then support those units as well. Each of the Emergency OSCs would be located in the respective annex buildings, and they would be located in the DCD designated area for the Technical Support Center. Member Armijo asked why the exclusion area boundary was larger for Unit 1 than for Units 2 and 3. The applicant responded that the design used the existing exclusion area for Unit 1, which was specified to be a mile, and the exclusion areas for Unit 2 and 3 were specified in the DCD which was roughly a half mile.

The applicant discussed data communication in accordance with the cyber security plan, and the data and communication for the Technical Support Center. Member Brown expressed concern about the potential communication with corrupted information. The applicant responded that if there were corrupted data, emergency response personnel would take care of the situation. There is a person in the main control room who is in continuous communications with the Technical Support Center. The person in the main control room would evaluate the operational actions and the data within the control room to ensure that the TSC is seeing both accurate and timely information.

The staff presented its review of the entire emergency plan with emphasis on Emergency Planning Zone size and the location of the TSC. The staff found the Emergency Planning to be acceptable.

The applicant responded to ACRS Action Item 63, which is a about calculations for hazards due to offsite chemicals. The applicant's analyses had shown that: 1) effects of explosions and flammable vapor clouds would not pose a threat to any safety-related systems, structures, or components, and 2) toxic vapor clouds would not exceed toxicity limits in the control room and would not pose a threat to control room operators. The staff provided its evaluation with the confirmatory analyses. Member Banerjee commented that any heavy liquefied gas would spread as a dense gas, and it will stay low to the ground. There were accidents of this nature where the vapor cloud had ignited, and it had moved without dispersion down topography. He asked if there was such topography that could take a dense gas cloud near the plant. The applicant responded that the topography is favorable at this site. The applicant explained that the plant is sitting on a cliff with a big head difference to the valley below. It would be very hard for a dense gas to move up the cliff towards the plant. Chairman Ray commented in agreement with Member Banerjee that the big elevation difference here favors the plant relative to any heavy gas. The staff also evaluated nearby accidents explosions. The staff used ALOHA as an analysis tool for liquid and gases and used RG 1.91 for the solid materials. The staff indicated that the pressure wave from all nearby explosives would not exceed 1 psi at safety-related SSCs. Dr. Banerjee commented that the ALOHA is the code from the applicant, and he asked the staff if other codes were used in the analysis. The staff responded that the staff used ALOHA with an independent input.

There were discussions about the HABIT Code, which the staff used for the confirmatory analysis for control room habitability. Member Banerjee had a concern with the validation of the code

related to its lacking heavy gas modeling and testing. Member Banerjee requested a reference for under what conditions HABIT had been verified. The staff came back in the second day and provided further discussion of the usage of the code HABIT. Member Armijo commented and Member Banerjee agreed that the basis for the staff's acceptance can't be limited the independent verification using HABIT; it has to include the analysis of the ALOHA code and the topography of the site.

The staff addressed ACRS Action Items from Section 2.5, which requested a comparison of the EPRI seismic source model used by the applicant with the most recent USGS model. The staff pointed out that the USGS maximum magnitude value is higher than EPRI values on average for source zones outside Charleston. Charleston seismic source models are similar in maximum magnitudes, recurrence interval, and source geometries. Dr. Hinze further asked about any reason why the FSAR should not include 2008 models as a reference. The staff responded that it wasn't ready when the applicant wrote the FSAR. The staff is planning to add the 2008 models to the SER but the applicant will not update its FSAR. During the discussion, the staff also presented some pictures from the excavation.

In the Chapter 1, Introduction and General Description of the Plant, presentation, the applicant introduced site location of the new planed Units 2 & 3 relative to the location of the existing Unit 1. The new unit site would be located about a mile south of the existing unit in Fairfield County. The overall site is about 26 miles from Columbia, SC, which is the nearest large population center. It also identified contractors and agents for this project, which included a construction contract consortium consisting of Westinghouse Electric Company and Shaw Group.

In its presentation for Chapter 5, 6, and 9, the application had a wet-bulb temperature exemption. The site-specific maximum safety non-coincident wet-bulb temperature was determined to be 87.3 F, based on the 1,000 year return value. This temperature is 1.2 F above the AP1000 DCD Tier 1 value. The applicant evaluated the impacts on the performance of systems and components and concluded that the performance is still acceptable with the increased wet bulb temperature. The staff presented its evaluation of this wet-bulb temperature departure and exemption. The evaluation covered the normal residual heat removal system, passive containment cooling system, habitability systems for main control room, component cooling systems, spent fuel pool cooling, service water system, and central chilled water system. The staff granted the exemption on the basis that the exemption does not have an adverse impact on the AP1000 standard plant design.

For the Chapter 3, the applicant and the staff presented site-specific supplements, which included wind and tornado loadings, flooding, turbine missiles, seismic design, and waterproofing material. The staff presented comparisons of the design ground motion response spectra, the hard rock high frequency spectra, and the certified seismic design response spectra.

For the Chapter 8, the applicant described a site-specific off-site power design and showed the transmission lines diagram and switchyard single-line diagram. The staff presented its evaluation and concluded that the applicant addressed the offsite power system adequately and provided sufficient information regarding the interfaces for standard design from the generic AP1000 DCD design.

For the Chapter 11, Radioactive Waste Management, the applicant discussed site-specific information, which includes the liquid radwaste system and wastewater system, gaseous waste management, effluent monitoring. Member Ryan asked a question regarding the use of the high-density polyethylene (HDPE) pipe. He expressed concern about the experience with using the

HDPE pipe and if the underground pipe will be monitored and how to detect leaks. The applicant responded with the description of the pipe layout configuration and the plan to use underground wells for monitoring. The same question was also asked to the Vogtle R COL representatives during the meeting. Member Ryan further asked how this pipe can hold up for sixty years and he recommended that this would be an area that needs to be addressed.

The topic of Loss of Large Area due to Fire or Explosion (LOLA) was discussed during this meeting. This topic included security related information and it was closed to the public.

Attachments

1. Sign-In Sheets
2. Meeting Agenda
3. Presentation Materials (open sections only)

Attachment 1

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON AP1000

January 10-11, 2011
Date

PLEASE PRINT

	NAME	AFFILIATION
1	Amy M. Monroe	SCE&G
2	ROBERT W HORTON	SCE+G
3	Stephen E. Summer	SCE&G
4	TIM SCHMIDT	SCE&G
5	Edward T. (Tim) Bonnette	SCE&G
6	Neil Haggerty	NuStart
7	William Hutchins	Westinghouse
8	Dan Patton	Bechtel (SCE&G)
9	Mary Richmond	Bechtel (SCE&G)
10	Angelos Findikakis	Bechtel (SCE&G)
11	James C LaBorde	SCE&G
12	Kevin Marsh	SCE&G
13	Ron Clary	SCE&G
14	STEVE BYRNE	SCE&G
15	Bob Williamson	SCE+G
16	MARK STELLA	Westinghouse
17	Thom Ray	Westinghouse
18	ED CUMMINS	"
19	Rolf Ziesing	Westinghouse
20	Mike Fayer	PNNL
21	JOHN DEBLASIO	WESTINGHOUSE
22	JASON BREHM	WESTINGHOUSE
23	~~RAJ~~ AL PAGES	~~[illegible]~~ P. SCE&G
24	TONY Pilo	Progress Energy
25	mike Melton	WEC
26		
27		
28		

Attachment 1

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON AP1000

January 10-11, 2011
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

	NAME	NRC ORGANIZATION-
1	JOHN BUDZYNSKI	NRO/DSRA/SRSB
2	VAUGHN THOMAS	NRO/DE/SCB1
3	MALCOLM PATTERSON	NRO/DSRA/SPRA
4	Justus Oldenburg	NRO/DSRA/SPRA
5	David Terao	NRO/DE/CIB1
6	Yong Li	NRO/DSER
7	Eduardo Sastre	NRO/DE/CIB
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Attachment 1

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON AP1000

January 10-11, 2011
Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

	NAME	NRC ORGANIZATION-
1	Joe Sebrosky	NRO/DNRL/NWE1
2	DON HABIB	NRO/DNRL/NWE1
3	Kenneth See	NRO/DSER/RHEB
4	GERRY L. STIREWALT	NRO/DSER/RGS2
5	Denise McGovern	NRO/DNRL/NWE1
6	DAN BARNHURST	NRO/DSER/RENV
7	Richard Raiom	NRO/DSER/RHEB
8	DAN BARSS	NSIR/DPR/NRLB
9	Jody Martin	OGC
10	Thomas Galletta	NRO/DNRL
11	ANTHONY BOWERS	NSIR/DPR
12	Christopher Cook	NRO/DSER
13	PAUL PIERINGER	NRO/DCIP/CQLP
14	Ned WRIGHT	NSIR/DPR-EP
15	George Cicotte	NRO/DCIP/CHPB
16	Ed Roach	NRO/DCIP/CHPB
17	Jill Caverly	NRO/DSER
18	JOHN RYCYNA	NSIR
19	ANTHONY MINGRIK	NRO/DNRL/NWE1
20	Ed Robinson	NSIR
21	David Brown	NRO/DSER
22	David Sisk	NRO/DSER
23	Nilesh Chokshi	NRO/DSER
24	RAJU PATEL	NRO/DCIP/CQVA
25	LARRY WHEELER	NRO/DSRA/SBP
26	Think Dinh	NRO/DSRA/SFPS
27	Raul Hernandez	NRO/DSRA/SBPA
28	Michelle Hayes	NRO/DSRA/SRSB

Attachment 2

Advisory Committee on Reactor Safeguards
Meeting of the Subcommittee on the
V.C. Summer Units 2 and 3 Combined License Application
Rockville, MD
January 10-11, 2011
Final

- Agenda -

Cognizant Staff Engineers: Weidong Wang (301-415-6279, Weidong.Wang@nrc.gov)
Peter C Wen (301-415-2832, Peter.Wen@nrc.gov)

January 10, 2011

Item	Topic	Presenter(s)	Time
1	Opening Remarks and Objectives	Harold B. Ray, ACRS	0830 – 0845
2	Applicant – Overview and Observations from onsite excavation activities and Section 2.4, Hydrology	Summer	0845 – 0910
3	Staff – Section 2.4, Hydrology	NRC	0910 – 0930
4	Applicant– Section 13.3, Emergency Planning, and Chapter 18	Summer	0930 – 1010
	Break		1010 – 1025
5	Staff – Section 13.3, Emergency Planning, and Chapter 18	NRC	1025 – 1100
6	Resolution of Action Items from July 2010 ACRS meeting and staff's observations from onsite excavation activities	Summer and NRC	1100 – 1145
7	Loss of Large Area due to fire or explosion (LOLA) CLOSED	Summer and NRC	1145 – 1200
	Lunch		1200 – 1300
8	Overview of application remaining presentations and FSAR chapter 1	Summer and NRC	1300 – 1330
9	Applicant Chapters 5, 6, and 9	Summer	1330 – 1350
10	Staff – Chapters 5, 6, and 9	NRC	1350 – 1415
	Break		1415 – 1430
11	Applicant Chapters 3, and 19 (PRA)	Summer	1430 – 1445
12	Staff – Chapters 3 and 19 (PRA)	NRC	1445 – 1500
13	Applicant – Chapters 8, and 10	Summer	1500 – 1520
14	Staff – Chapters 8 and 10	NRC	1520 – 1540
15	Committee Discussion	Harold B. Ray, ACRS	1540 – 1600
	Adjourn		1600

Notes:
Presentation time should not exceed 50% of the total time allocated for a specific item.
Number of copies of presentation materials to be provided to the ACRS - 35.

Attachment 2

January 11, 2011

Item	Topic	Presenter(s)	Time
1	Opening Remarks and Objectives	Harold B. Ray, ACRS	0830 – 0835
2	Applicant – Chapters 11, 12, and 13 (without emergency planning)	Summer	0835 – 0850
3	Staff – Chapters 11, 12, and 13 (without emergency planning)	NRC	0850 – 0910
4	Applicant – Chapters 15, 16, and 17	Summer	0910 – 0925
5	Staff – Chapters 15, 16 and 17	NRC	0925 – 0940
6	Committee Discussion	Harold B. Ray, ACRS	0940 – 1000
	Adjourn		1000

Notes:
Presentation time should not exceed 50% of the total time allocated for a specific item.
Number of copies of presentation materials to be provided to the ACRS - 35.

Attachment 3

SCE&G • Santee Cooper
Shaw • Westinghouse Electric Company

VC Summer Units 2 and 3 Overview and Observations from Excavation Activities

Stephen A. Byrne – Executive VP – Generation
Alfred M. Paglia - Manager Licensing - New Nuclear Deployment

Attachment 3

Attachment 3

Attachment 3


January
2011

Attachment 3

SCE&G • Santee Cooper
Shaw • Westinghouse Electric Company

VC Summer Units 2 and 3 FSAR Section 2.4

Steve Summer

SCANA Services – Supervisor
Environmental Services

FSAR Section 2.4 Hydrologic Engineering

- DCD Incorporated By Reference
- No Exemptions Requested
- 1 Administrative Departure – 2.0-1 pertaining to section numbering to align with RG 1.206 and facilitate NRC review

Major Items of Interest

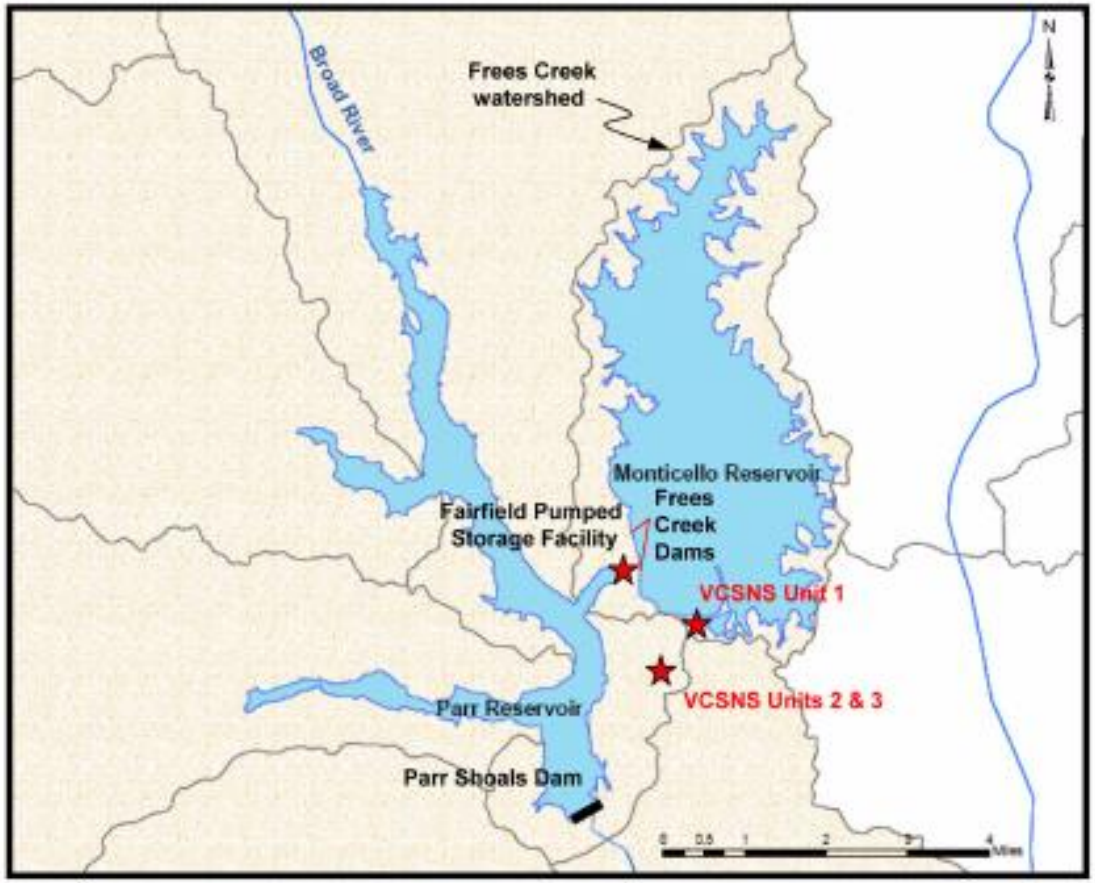
- 6 COL Information Items Addressed
  - 2.4-1 Hydrological Description
  - 2.4-2 Floods
  - 2.4-3 Cooling Water Supply
  - 2.4-4 Groundwater
  - 2.4-5 Accidental Release of Liquid Effluents into Ground and Surface Water
  - 2.4-6 Flood Protection Emergency Operation Procedures

Major Items of Interest

- COL Item 2.4-1 Hydrological Description - **Describe major hydrologic features on or in the vicinity of the site.**

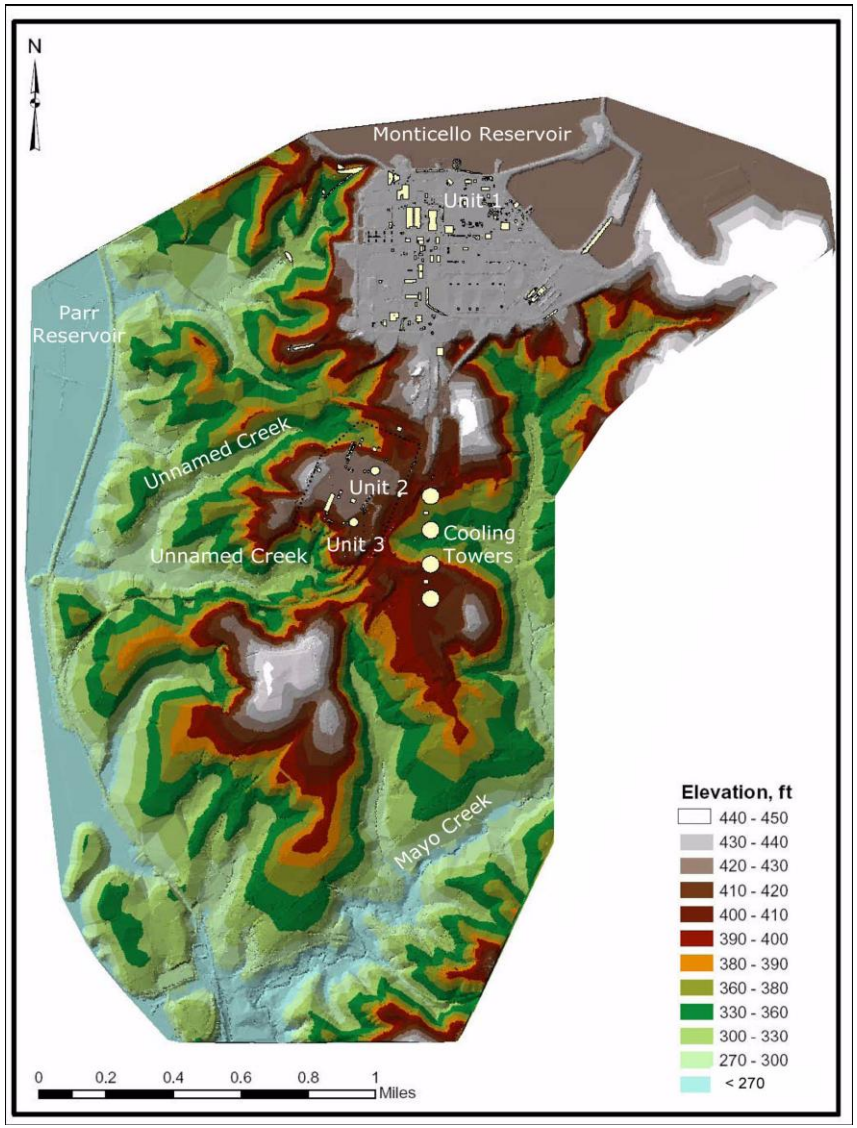
Attachment 3

Major Surface Water Hydrologic Features



Data sources:
1. USGS DEM files for Jenkinsville and Salem Crossroads
2. Area survey data from Glenn Associates Surveying, Inc
3. Recent satellite image of the Monticello Reservoir area

Site Topography



Major Items of Interest

- COL Item 2.4-2 Floods - **Address site-specific information on historical flooding and potential flooding factors, including the effects of local intense precipitation.**
  - No risk to Safety-Related Systems, Structures, or Components (SR SSCs) from flooding.
    - Probable Maximum Flood level is more than 100 feet below site grade
    - Site is not susceptible to surges, seiches and tsunami.

Major Items of Interest

- COL Item 2.4-3 Cooling Water Supply - **Address the water supply sources to provide makeup water to the service water system cooling tower.**
  - The Broad River and Monticello Reservoir are used as the cooling water makeup source (non-safety related).
  - Ice effects are highly unlikely.
  - The Broad River is adequate for non-safety uses even during low-flow conditions.

Major Items of Interest

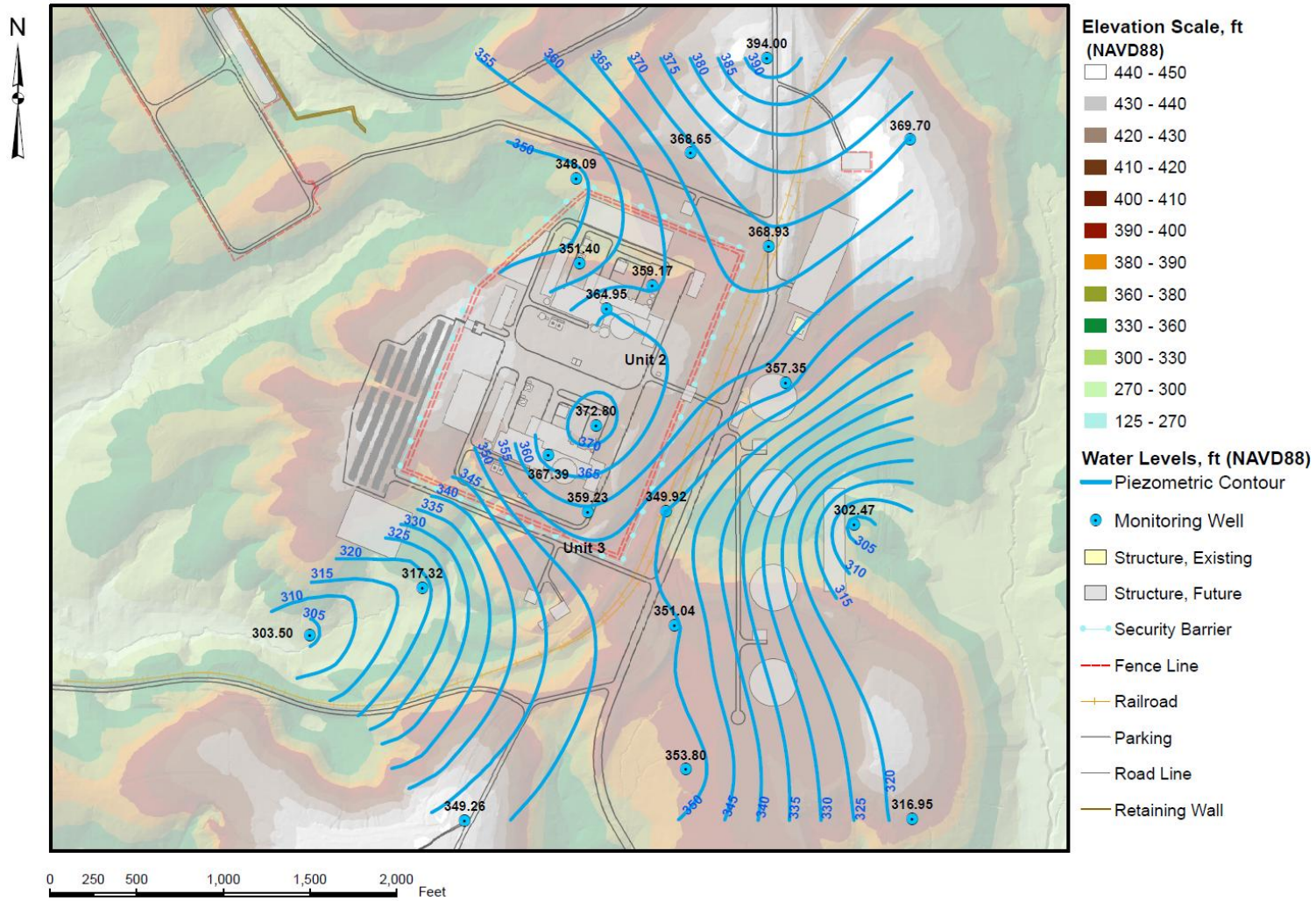
- COL Item 2.4-4 Groundwater - **Address site-specific information on groundwater**.
  - There are no plans to use local groundwater for construction or operation of VCSNS Units 2 and 3.

Major Items of Interest

- COL Item 2.4-4 Groundwater (Cont'd)
  - Units 2 and 3 are located on a ridgetop. Piezometric contour maps indicate that groundwater from the ridgetop flows away from the site.

Attachment 3

Major Items of Interest



Major Items of Interest

- COL Item 2.4-4 Groundwater (cont'd)
  - Design plant grade elevation is 400 feet NAVD88.
  - The maximum allowable groundwater level is 398 feet NAVD88 (AP1000 DCD).
  - The maximum expected groundwater level is 380 feet NAVD88 (20 feet below the plant grade elevation), well below DCD value.

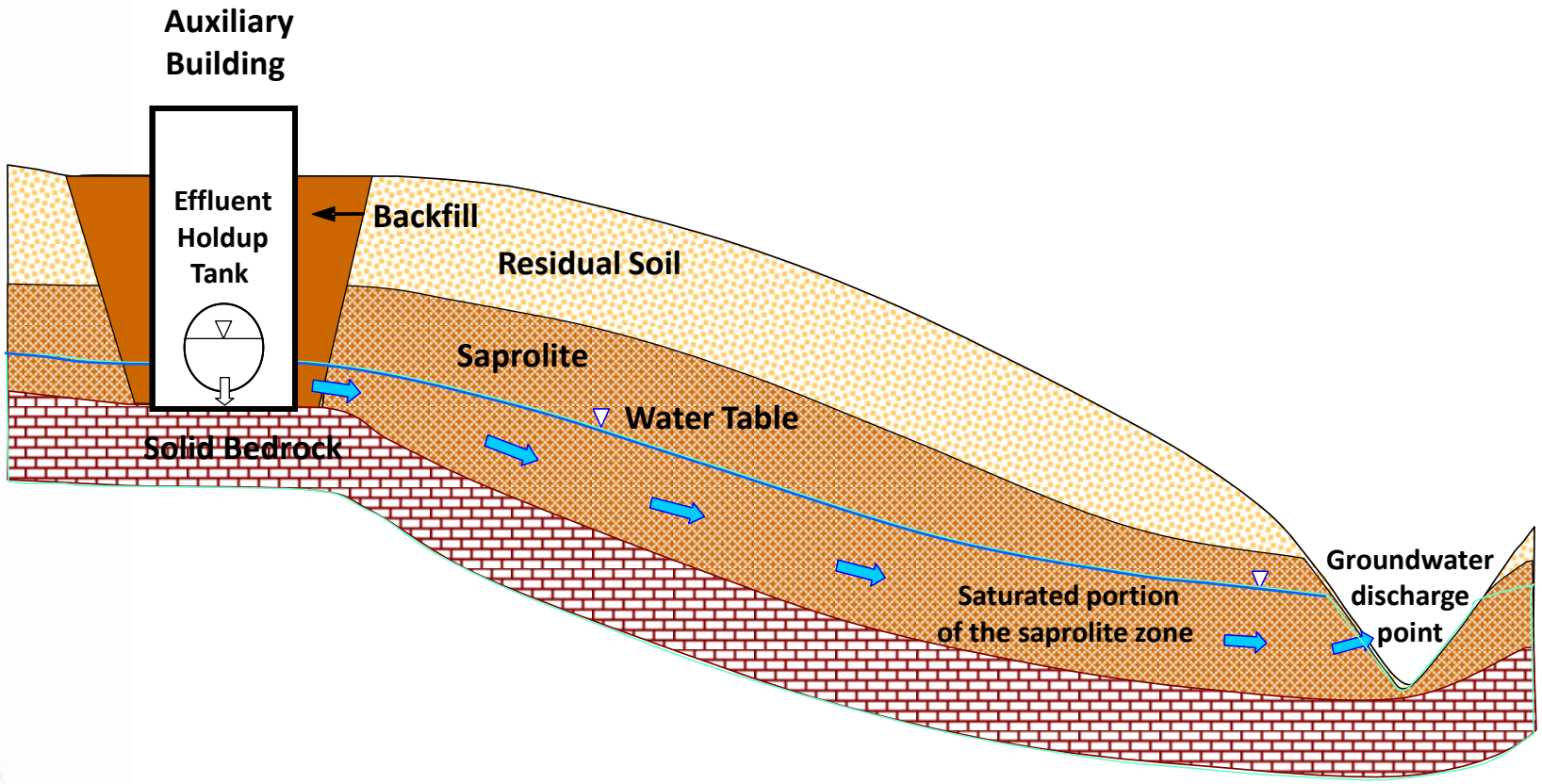
Major Items of Interest

- COL Item 2.4-5 - Accidental Release of Liquid Effluents into Ground and Surface Water - **Address site-specific information on the ability of the ground and surface water to disperse, dilute, or concentrate accidental releases of liquid effluents. Also address the effects of these releases on existing and known future use of surface water resources.**

Major Items of Interest

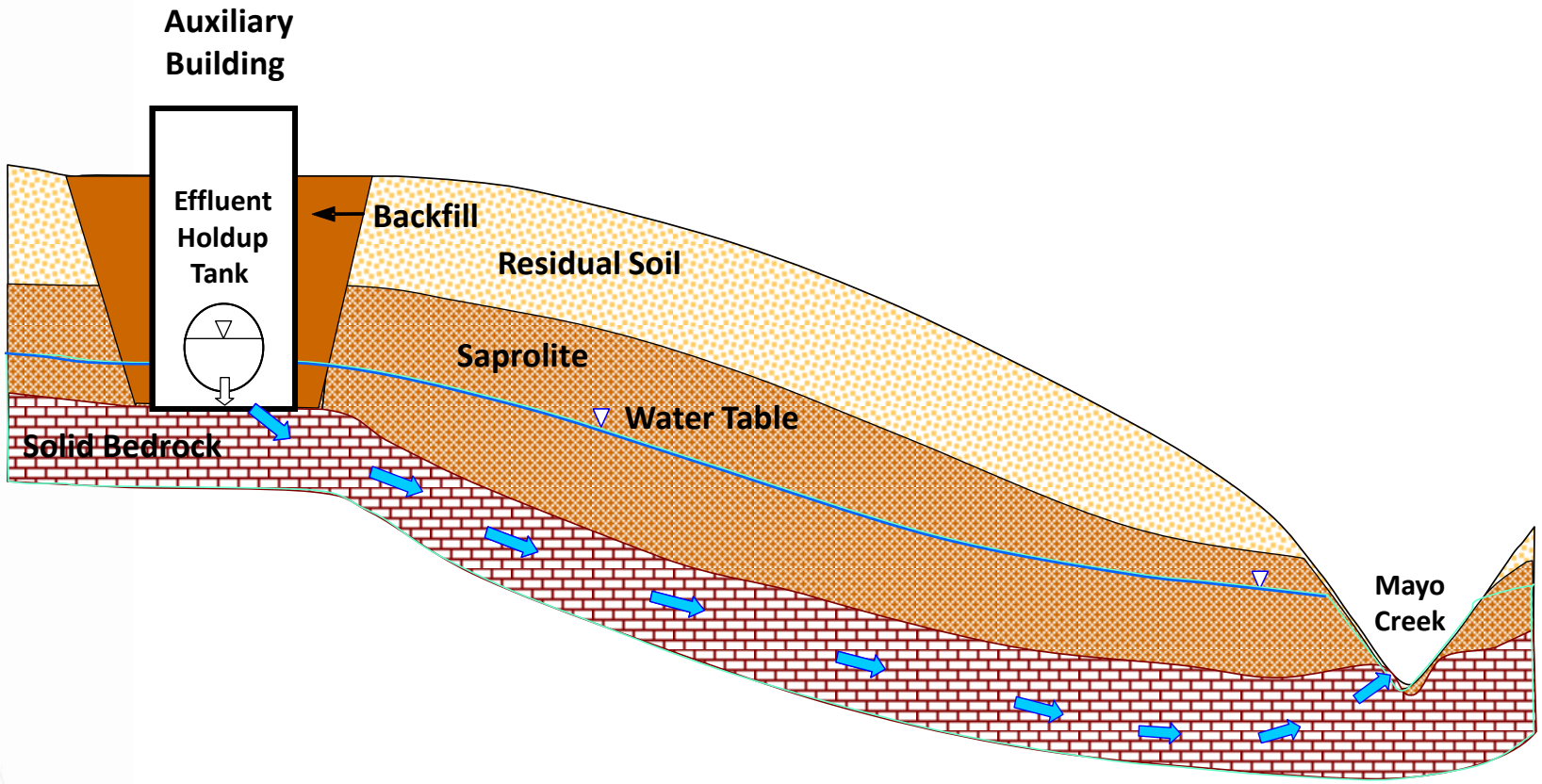
- COL Item 2.4-5 - Accidental Release of Liquid Effluents into Ground and Surface Water (cont'd)
  - Evaluation shows that an accidental liquid release of effluents in groundwater would not exceed 10 CFR Part 20 limits.
  - Three conceptual flow transport models (one saprolite and two bedrock) are presented.

Major Items of Interest



Saprolite Pathway (conceptual)

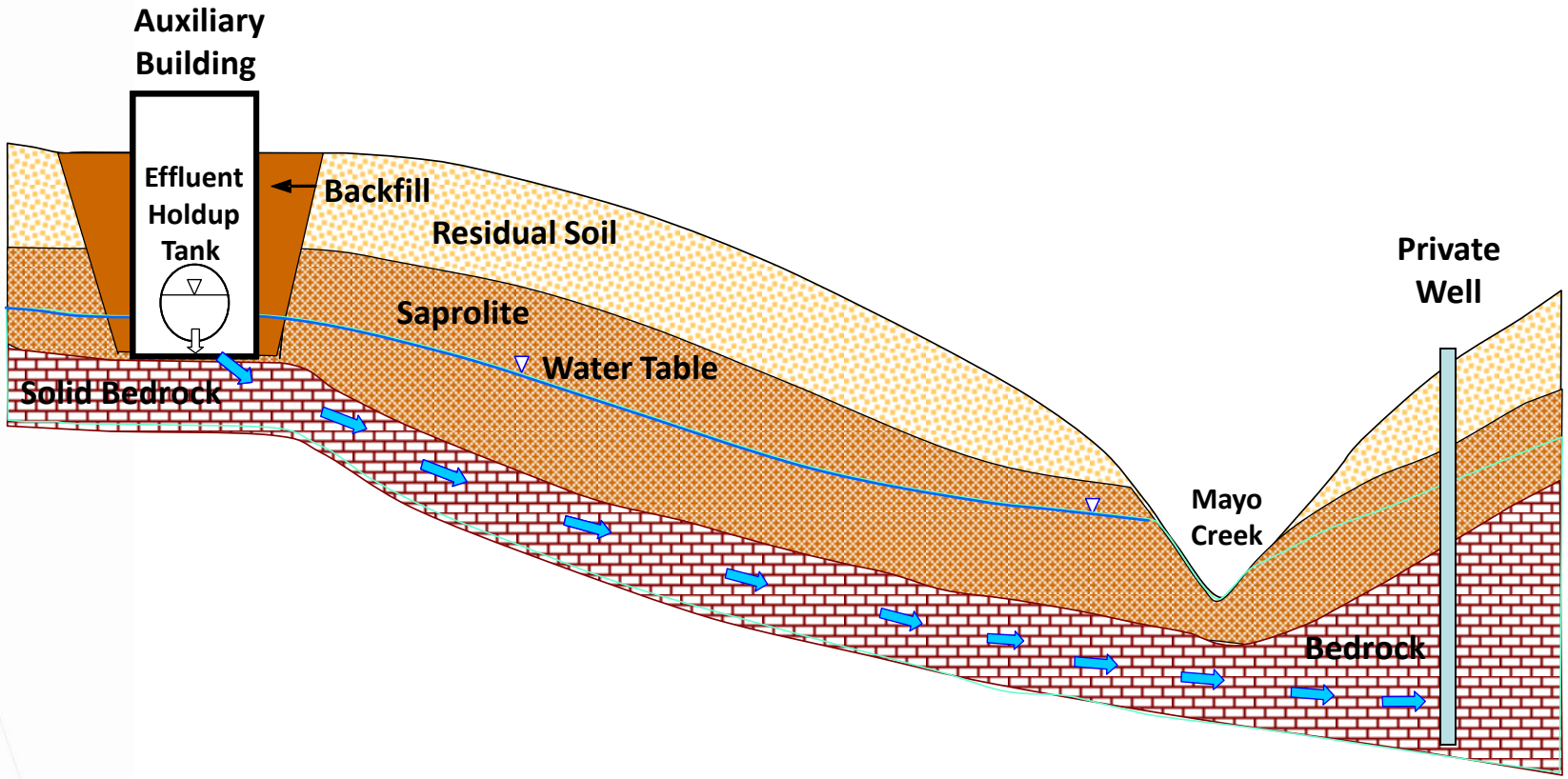
Major Items of Interest



Bedrock Pathway to Broad River or Mayo Creek

(conceptual)

Major Items of Interest



Bedrock Pathway to the site boundary below Mayo Creek (conceptual)

Major Items of Interest

- COL Item 2.4-6 Flood Protection Emergency Operation Procedures - **Address any flood protection emergency procedures required to meet the site parameter for flood level.**
  - Since the SR SSCs at Units 2 and 3 are not subject to flooding, no additional flood protection measures and no emergency procedures are required.

Major Items of Interest

- RAIs
  - 2 questions on flooding
  - 6 questions on groundwater
  - 14 questions on accidental release of radioactive liquid effluents in ground and surface waters

  - All questions have been answered and are considered to be resolved.

Questions?

Attachment 3

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

Summer Units 2 and 3 COL Application Review

Section 2.4

Hydrologic Engineering

January 10 - 11, 2011

Staff Review Team

- Technical Staff
  - Ken See
  - Daniel Barnhurst
  - Steven Schaffer
  - Lance Vail, Pacific Northwest National Lab (PNNL)
  - Lyle Hibler, PNNL
  - Mike Fayer, PNNL
- Project Management
  - Joe Sebrosky

Floods (FSAR Sections 2.4.1 through 2.4.5, and 2.4.7 through 2.4.10)

- The staff reviewed various flooding mechanisms to determine the site characteristics related to design-basis flood and required flooding protection.
- Specific items of interest:
  - The applicant identified the flood caused by local intense precipitation as the design-basis flood.
  - The staff has identified Confirmatory Items 2.4.2-1 which specifies in future revisions to the FSAR channel maintenance procedures be described and cross-section maps used in the analysis be provided .
  - The staff analyzed the flood potential due to a postulated failure of the Monticello Reservoir berm; and confirmed applicant's determination of local intense precipitation as the DBF.

Attachment 3

FSAR Sections 2.4.4: Major Hydrologic Surface Water Features

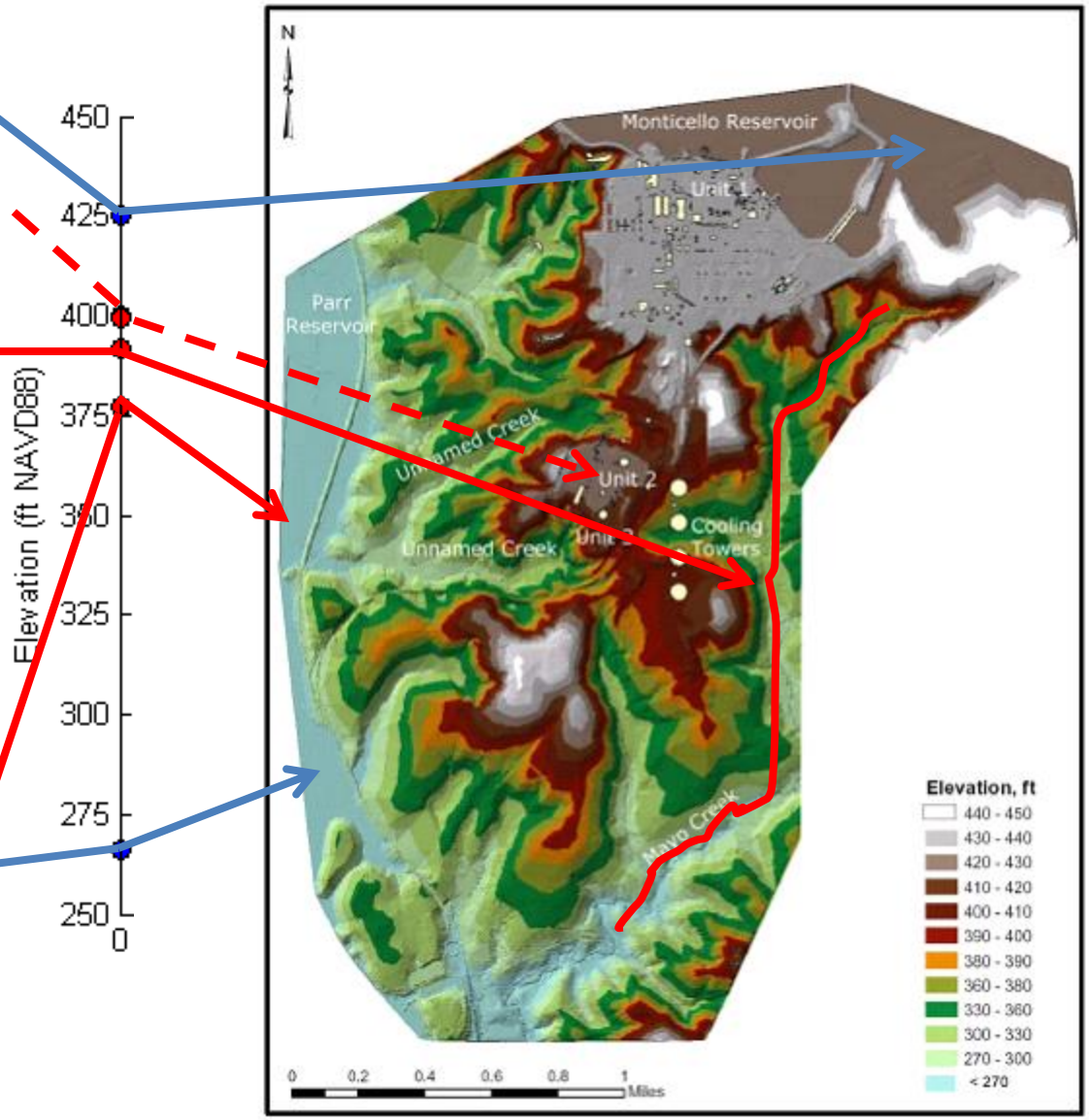
Monticello Reservoir Max Operating Pool Elevation

Site Grade (400) & Local Intense Precipitation Peak Elevation (399.4)

Monticello Reservoir Dam Breach Peak Elevation Near Site

Upstream Broad River Dam Breach Peak Elevation at Parr Shoals

Parr Shoals Max Operating Pool Elelvation



Effects of Local Intense Precipitation (FSAR Section 2.4.2.4.3)

The applicant identified the flood caused by local intense precipitation as the design-basis flood. NRC staff confirmed this determination.

- Specific items of interest:
  - HEC-RAS was used to estimate peak flood elevations.
  - Staff determined that peak elevations from the postulated breach would not exceed the site grade elevation.
  - Staff requested in RAI 2.4.2-1 that the applicant provide a description of the program to ensure that drainage channels remain clear as a result of staff findings on the site drainage sensitivity to channel roughness after local intense precipitation events
  - Staff requested in RAI 2.4.13-14 a map of the modeled cross-section to support the local intense precipitation analysis be included in the FSAR
  - Items are being tracked for inclusion in future revision of the FSAR as Confirmatory Item 2.4.2-1

Potential Dam Failures (FSAR Section 2.4.4)

- The staff postulated a dam failure scenario in addition to those discussed in the FSAR by the applicant
- Specific items of interest:
  - The berm between Monticello Reservoir and Mayo Creek was postulated to fail and results flow into Mayo Creek.
  - Dam Safety Office guidance was used to develop estimates for dam breach peak flows.
  - HEC-RAS was used to estimate peak flood elevations.
  - Staff determined that peak elevations from the postulated breach would not exceed the site grade elevation.

FSAR Section 2.4.10: Flooding Protection Requirements

- The staff reviewed the characteristics of the design-basis flood for any required flooding protection.
- The NRC staff has established the local intense precipitation event as the DBF and as discussed in Section 2.4.2 of the SER, the staff determined that flood protection is not required.

FSAR Section 2.4.12: Groundwater

- The staff reviewed the hydrogeological characteristics of the site.
  - Applicant measured characteristics and properties to support groundwater conceptual models and estimate direction and velocity of potential radioactive contaminants.
  - Applicant determined maximum groundwater level would remain below the DCD requirement
- Specific items of interest:
  - Staff reviewed the characteristics and properties of the proposed site as described by the applicant.
  - Staff concluded that hydrogeological characterization is sufficient to support both the groundwater conceptual model and the site characteristic for maximum groundwater elevation based on supplemental information to be included in the FSAR
  - Staff established Confirmatory Item 2.4.12-1 to verify information is included in next revision of FSAR Section 2.4

FSAR Section 2.4.13: Accidental Releases of Radioactive Liquid Effluent in Groundwater and Surface Water

- The staff reviewed postulated accidental release from the radwaste management system and its potential effects on groundwater and surface water.
  - Applicant evaluated the ability of the groundwater and surface water environment to delay, disperse, dilute, or concentrate liquid effluent.
  - Applicant described the effects of postulated releases on known and likely future uses of water resources.
- Specific items of interest:
  - Staff reviewed the postulated release and pathway analysis methodologies and determined they were acceptable
  - Staff examined the results and determined that the concentrations were below the acceptance criteria in Branch Technical Position 11-6.
- The staff's review of the FSAR Section has been completed

Attachment 3

SCE&G • Santee Cooper
Shaw • Westinghouse Electric Company

VC Summer Units 2 and 3
FSAR Section 13.3
Emergency Planning

Tim Bonnette

SCE&G – Emergency Preparedness

Presentation Overview

- DCD Departure
- Plan Design
- Site Layout
- Command and Control
- Emergency Facilities
- Emergency Response
- Emergency Planning Zone
- Offsite Education and Alerting

DCD Departure

- VCS DEP 18.8-1 – Locations of the Technical Support Center (TSC) and Operational Support Center (OSC)
  - TSC will be located in the New Nuclear Operations Building
  - Each OSC for Units 2 & 3 will be located in its respective Annex Building, in the area designated as the DCD TSC.

Emergency Plan Design

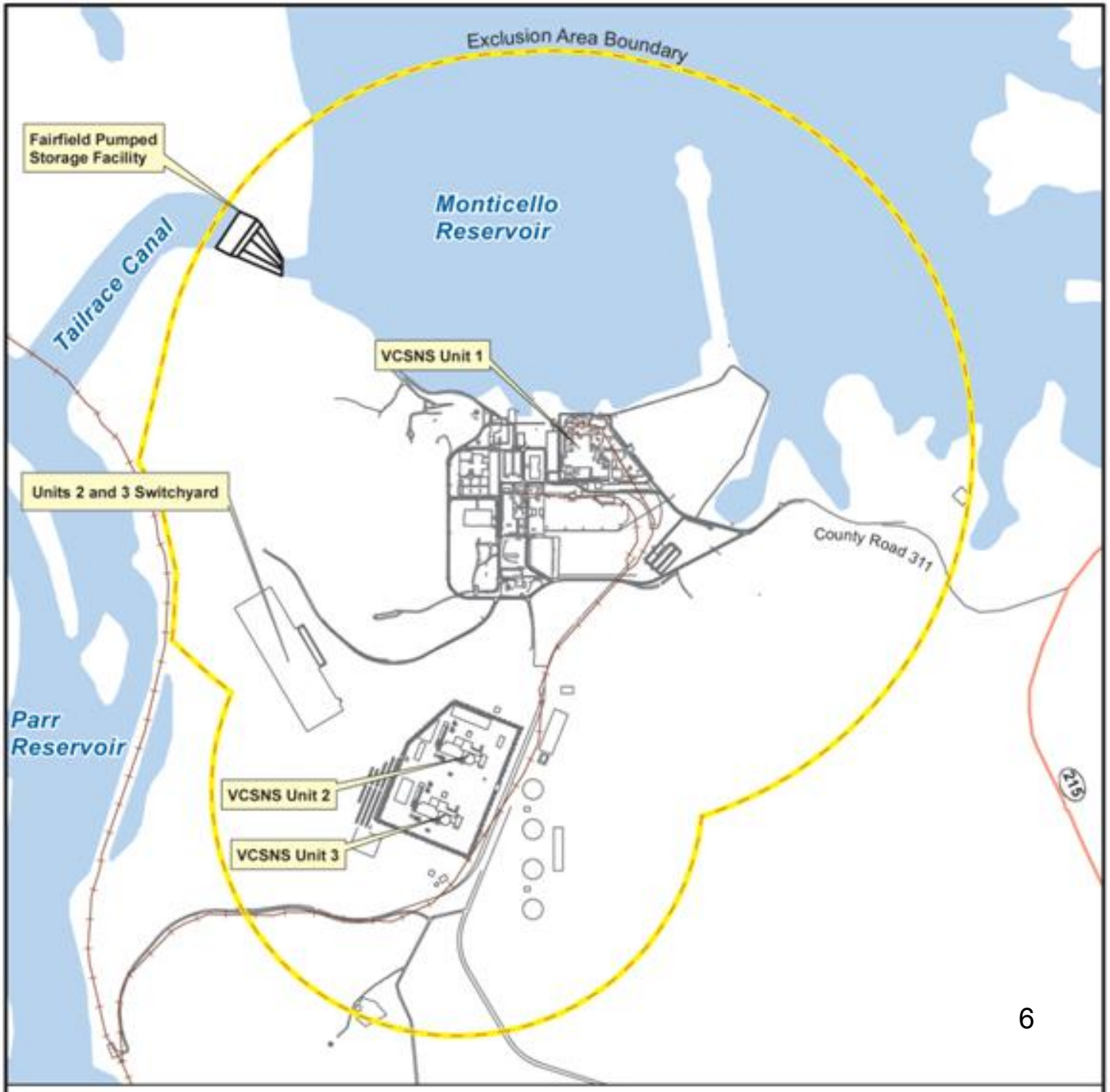
- Single plan for all three Units
  - In accordance with NUREG-0654
- Developed in accordance with:
  - NUREG-0654/FEMA-REP-1 Rev 1
  - 10 CFR 50.47
  - 10 CFR 50 Appendix E
- Emergency Action Level (EALs) developed in accordance with:
  - NEI 07-01 Rev 0

VCS Site Layout

- Single Nuclear Exclusion Area
  - Two points of ingress and egress into the Nuclear Exclusion Area
    - South of the Units
    - East of the Units
- Dual Protected Areas
  - Unit 1
  - Units 2 & 3

Attachment 3

Site Map



Command and Control

- Activation of the Emergency Response Organization (ERO) at an Alert or higher classification
  - Each emergency facility has a element of command and control
- Declared emergency involving a single Unit
  - The affected Unit's Control Room has the lead
- Declared emergency involving the entire Site
  - Unit 1 Control Room is the lead Control Room
- Declared emergency involving Units 2 & 3 only
  - Unit 2 Control Room is the lead Control Room

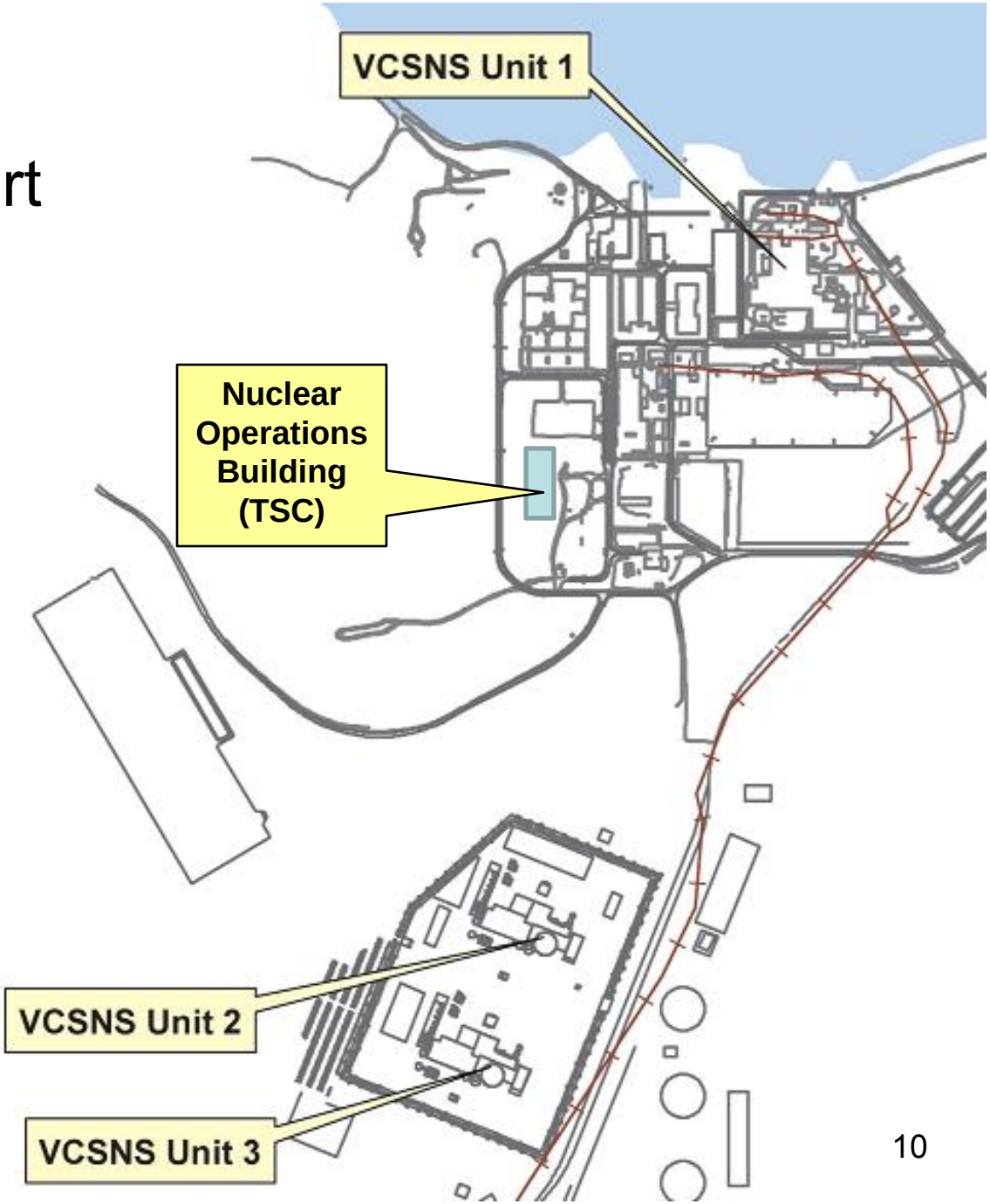
Emergency Facilities

- 3 Control Rooms
  - Unit 1 (existing)
  - Units 2 & 3 – located per DCD
- 3 Operational Support Centers (OSC)
  - Unit 1 (existing)
  - Units 2 & 3 – located in the respective Annex Building on the DCD 117’-6” Elevation

Emergency Facilities

- Technical Support Center (TSC)
  - Common for all three Units
  - Meets the requirements of NUREG-0696, with exception of being adjacent to the Control Rooms
  - Data and communication links between each Unit and the TSC are in accordance with the Cyber Security Plan

Technical Support Center

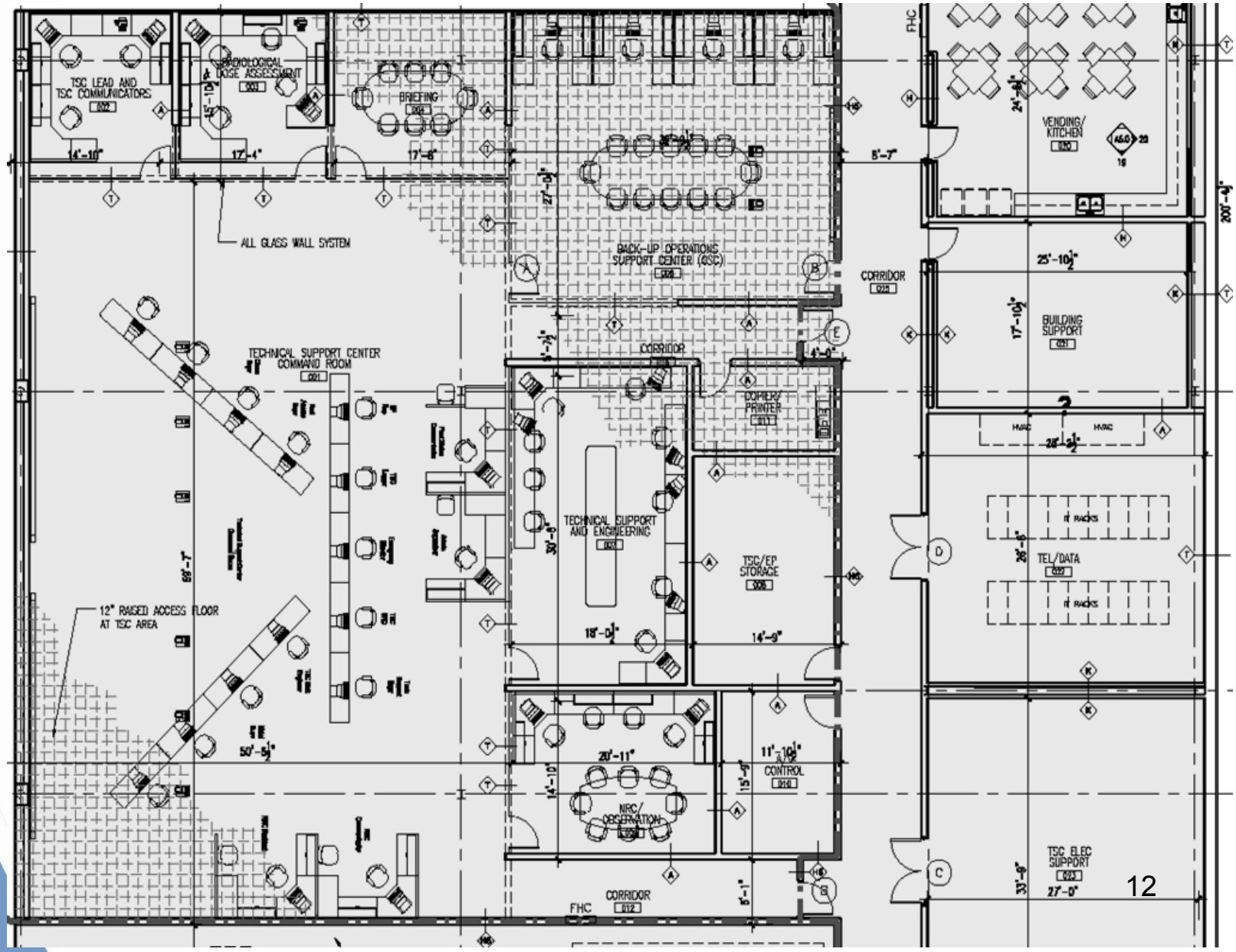


Emergency Facilities

- TSC (continued)
  - Incorporates human factors engineering (HFE) to support emergencies involving one, two, or three Units
    - Centralized Command Area
    - Adjacent support areas
  - ERO positions support continuous collaboration with the affected Control Room(s)

Attachment 3

Technical Support Center Layout



Emergency Facilities

- Emergency Operations Facility (EOF)
  - Existing
  - Common for all three Units
  - Located offsite, outside the Emergency Planning Zone (EPZ)
- Joint Information Center (JIC)
  - Existing
  - Common for all three Units
  - Located offsite, outside the EPZ

Attachment 3

EOF and JIC Facilities

Emergency Response

- Unusual Event Classification
  - Lead Control Room Shift Supervisor becomes the Interim Emergency Director (IED)
  - Supported by:
    - Shift Staffing from both the affected and unaffected Units
    - Additional staffing called in at the IED's discretion
  - All activities are controlled through the Control Room or by assigned personnel
  - Escalation to a higher classification requires activation of the Emergency Response Organization (ERO)

Emergency Response

- Alert, Site Area Emergency, and General Emergency Classifications
  - The Control Room (SRO) is the lead for operational plant monitoring and operational control
  - The TSC takes the lead in the onsite evaluations and decision making for mitigation strategies, in collaboration with the Control Room(s)
  - The EOF takes the overall Command and Control and is the lead facility for classifying an event, recommending Protective Actions, and notifying the offsite authorities

Emergency Response

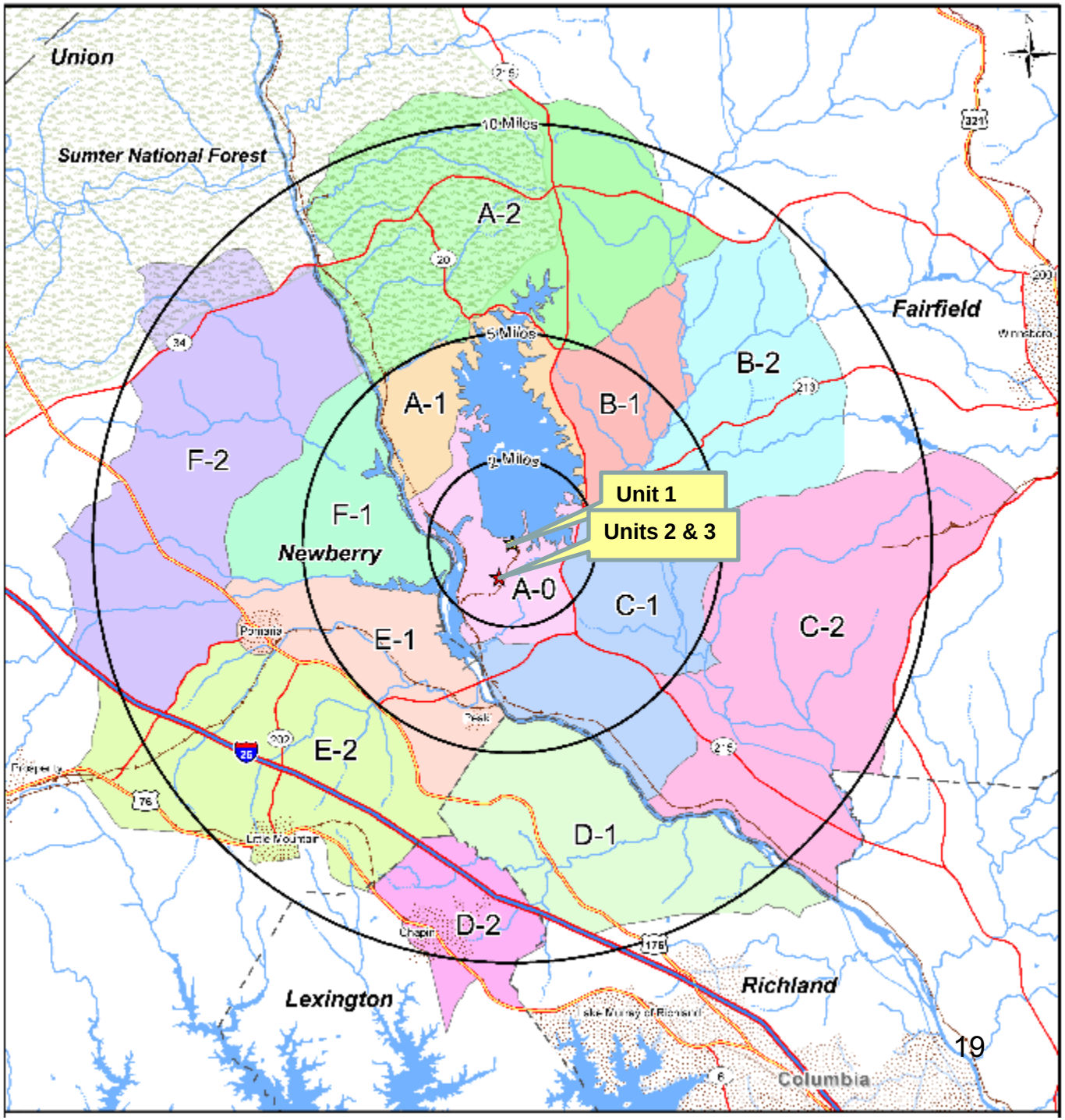
- Alert, Site Area Emergency, and General Emergency Classifications (continued)
  - The OSC provides the support personnel to implement the in-plant mitigation strategies and conduct onsite evaluations to protect public health and safety
  - The JIC provides the media interfaces needed to ensure the public is kept up to date with the event and mitigation progression

Emergency Planning Zone (EPZ)

- EPZ boundaries remain the same
- Agreed upon by the State of SC and the risk counties (Fairfield, Lexington, Newberry, & Richland)
- Reviewed and accepted by FEMA

Attachment 3

EPZ Map



Offsite Education

- Annual Calendar Distribution
  - Details actions and guidance for members of the public
  - Distributed to all residents and businesses within the EPZ, as well as to all site employees
  - Includes self addressed and postage paid cards for residents with special needs
- Emergency Responder Training
  - Basic radiological training is provided to all first response agencies within the risk counties and selected State agencies

Offsite Alerting

- VCS Notifications and Alerting
  - Emergency Notification Form
  - Alert and Notification System
- State and Local Alerting
  - Back-up Route Alerting
  - Emergency Alert System

Questions?

Attachment 3

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

Virgil C. Summer Units 2 and 3 COL Application Review

ASE Section 13.3 and Chapter 18

Emergency Planning, and
Human Factors Engineering

January 10-11, 2011

Staff Review Team

- Technical Staff
  - **Ned Wright,** Section 13.3
  - **Paul Pieringer,** Chapter 18

- Project Managers
  - **Denise McGovern,** Section 13.3
  - **Anthony Minarik,** Chapter 18

Emergency Planning

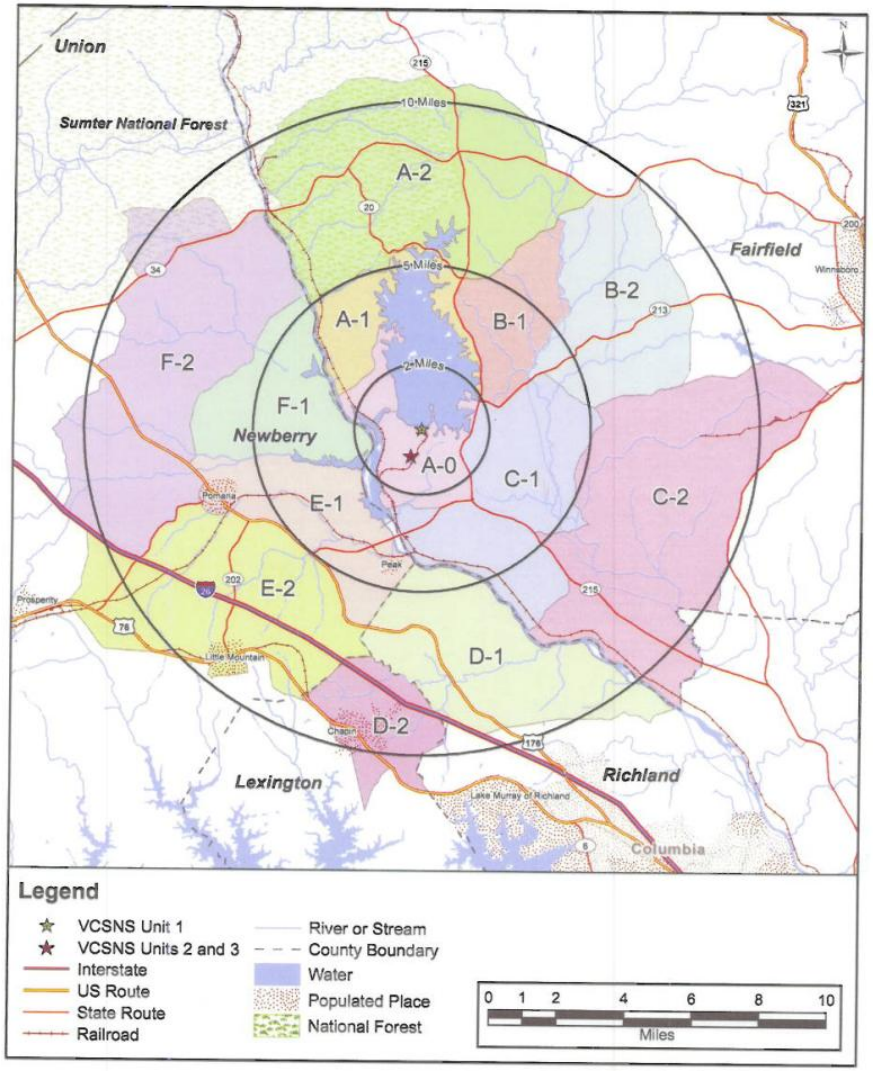
- EPZ
  - Applicant has proposed an EPZ for Units 2/3 that is the same for Unit 1
  - Reviewed and approved by the State of South Carolina and 4 Risk Counties prior to COL submittal
  - FEMA inquired as to whether the EPZ needed to be expanded based on the new reactor siting

Technical Support Center Location

- Consolidated TSC for all 3 units
  - Distance
    - Transit time from any MCR is 10-15 minutes
    - Compensated by enhanced communications capabilities
  - Adequate Capability
    - Demonstration of capability by an EP-ITAAC

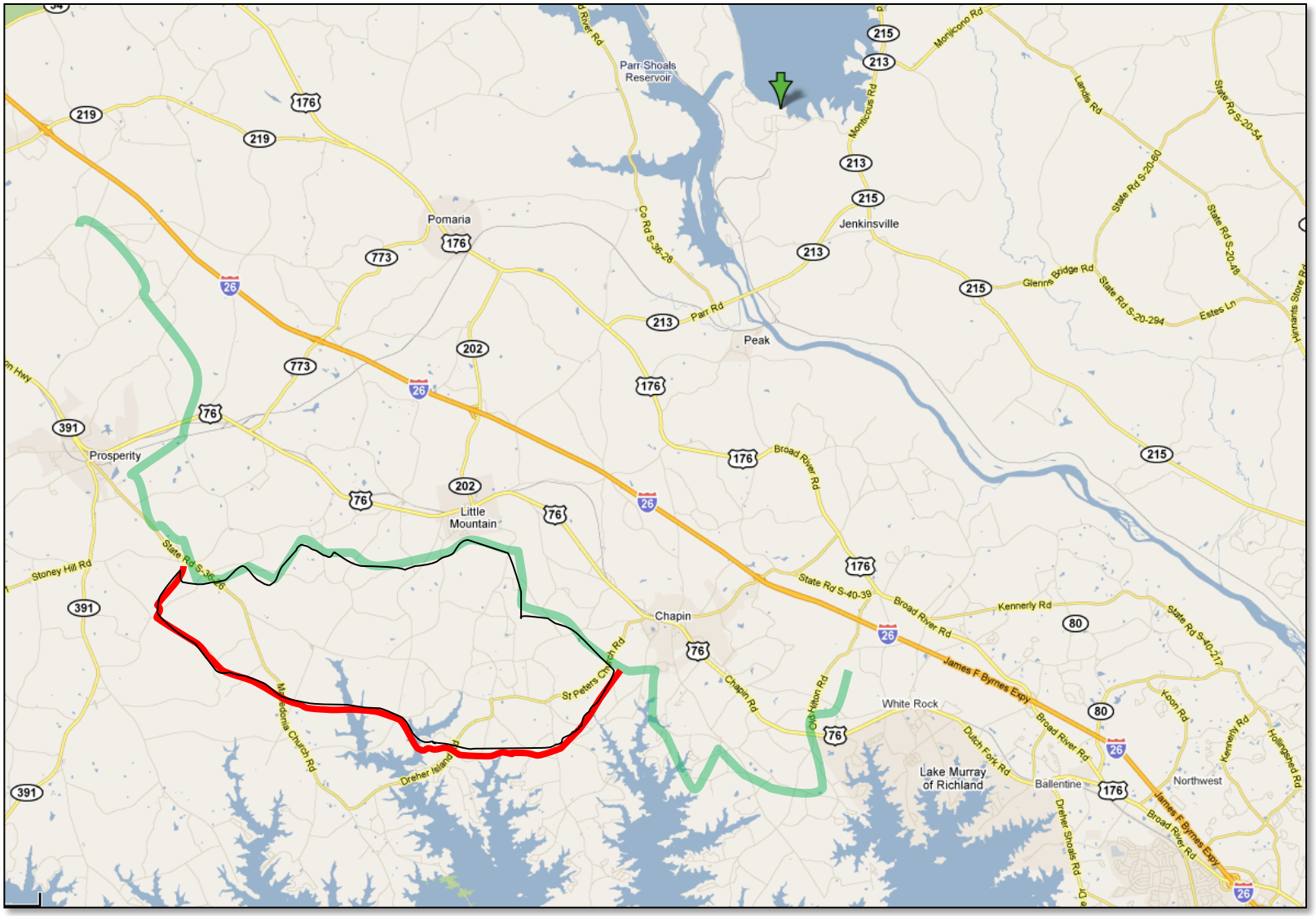
Attachment 3

VC Summer EPZ



Attachment 3

VC Summer EPZ



Supplemental Information

- The VCSNS Emergency Plan describes dedicated and diverse communications capabilities between the control rooms, TSC, OSC, and the EOF. These dedicated communications links include:
  - phone link for the Affected Unit to dispatch OSC teams between the OSC, TSC, and Control Room.
  - phone link for use by the ED, EPM, and Shift Supervisor/EPOS between the Affected Unit Control Room, the TSC, and the EOF.
  - phone link for transmission of technical data between the TSC, Affected Unit Control Room, and the EOF.
  - phone link to discuss mitigating activities and priorities between the TSC and EOF.
  - Station telephone line that is a communication link between activated facilities.
- The phone links in the station have diverse and back-up power supplies

Attachment 3

Summer FSAR Chapter 18

Human Factors Engineering (HFE)

FSAR Section	Site-Specific Evaluations
18.1 Overview	• None*
18.2 HFE Program Management	• VCS COL 18.2-2, Location of the Emergency Operations Facility
18.3–18.7	• None*
18.8 Human-System Interface Design	• VCS DEP 18.8-1, Location of the Technical Support Center (TSC) and Operational Support Center (OSC)
18.9–18.14	• None*

* This section is entirely IBR or IBR/standard.

Attachment 3

VC Summer Units 2 and 3 Action Item 63

Amy M. Monroe – Licensing
New Nuclear Deployment

FSAR 2.2.3 Evaluation of Potential Accidents

- ACRS requested calculations on hazards due to offsite chemicals were provided (ML103140717).

- Hazard scenarios were evaluated for each accident category identified in RG 1.206, including hazards from explosions, flammable vapor clouds (delayed ignition), and toxic chemicals from nearby transportation and industrial facilities.

FSAR 2.2.3 Evaluation of Potential Accidents

- Analyses showed that effects of explosions and flammable vapor clouds would not pose a threat to any safety-related systems, structures, or components.
- Analyses showed that toxic vapor clouds would not exceed toxicity limits in the control room and would not pose a threat to control room operators.

Questions?

Attachment 3

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

V. C. Summer Units 2 and 3 COL Application Review

Action Item 63
(Staff confirmatory calculations of offsite chemical releases)

January 10 -11, 2011

Staff Review Team

- Technical Staff
  - David Sisk, Siting and Accident Consequences Branch
  - Shie-Jeng Peng, Containment & Ventilation Branch
- Project Managers
  - Don Habib and Joe Sebrosky

Section 2.2.3, Evaluation of Nearby Accidents – Toxic Chemicals

- Brief ACRS on confirmative calculation results on the impact on control room habitability due to potential releases of offsite chemicals
- Staff evaluated chemical hazards stored or transported within 5 miles of the site
- Staff used ALOHA to determine safe distances
- Distance to the control room at ground level was less than the calculated safe distances
- Three site-specific chemicals could exceed IDLH at the Control Room Intake:
  - 28% ammonium hydroxide (Unit 1)
  - Cyclohexylamine (Norfolk Southern rail)
  - Chlorodifluoromethane (Norfolk Southern rail)

Section 2.2.3, Evaluation of Nearby Accidents – Explosions

- Max. probable solid boxcar cargo = 132,000 lbs (RG 1.91)
- TNT equivalence = 1 for non-munition explosives (RG 1.91)
- One boxcar is evaluated because pressure waves from subsequent explosions are not cumulative
- Exceptions include certain exotic materials (nitroglycerine)
- Rail cargos near this site do not normally contain munitions or exotic materials
- Staff evaluated solid materials using RG 1.91
- Staff evaluated liquid and gases using the ALOHA
- Staff found that the pressure wave from all nearby explosives would not exceed 1 psi at safety-related SSCs

VCS COL 6.4-1, Concentrations of Site-Specific Chemicals, Staff Confirmative Calculation Results (HABIT)

- Staff performed a more detailed analysis for the 3 toxic chemicals using HABIT.

Chemical	MCR Concentration	IDLH Limit
28% Ammonium hydroxide (VCSNS Unit 1)	68 ppm	300 ppm
Cyclohexylamine (Offsite railcar)	4 ppm	10 ppm
Chlorodifluoromethane (Offsite railcar)	357 ppm	1,200 ppm

Attachment 3

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Presentation to the ACRS Subcommittee

V.C. Summer Nuclear Station Units 2 and 3 COL Application Review

Advanced Safety Evaluation Section 2.5
Geology, Seismology, and Geotechnical Engineering

January 10, 2011

Staff Review Team

- Technical Staff
  - Dr. Clifford Munson, Senior Level Advisor and Seismologist
  - Dr. Gerry L. Stirewalt, Senior Geologist
- Project Management
  - Joe Sebrosky

Overview

- Section 2.5
  - Topics of Interest
    - Action item from July 2010 ACRS meeting to compare EPRI seismic source model used by applicant with most recent USGS model
    - Field observations by NRC geologists on geologic mapping of the Unit 2 excavation for assessing the presence of potential tectonic features (August 2010)

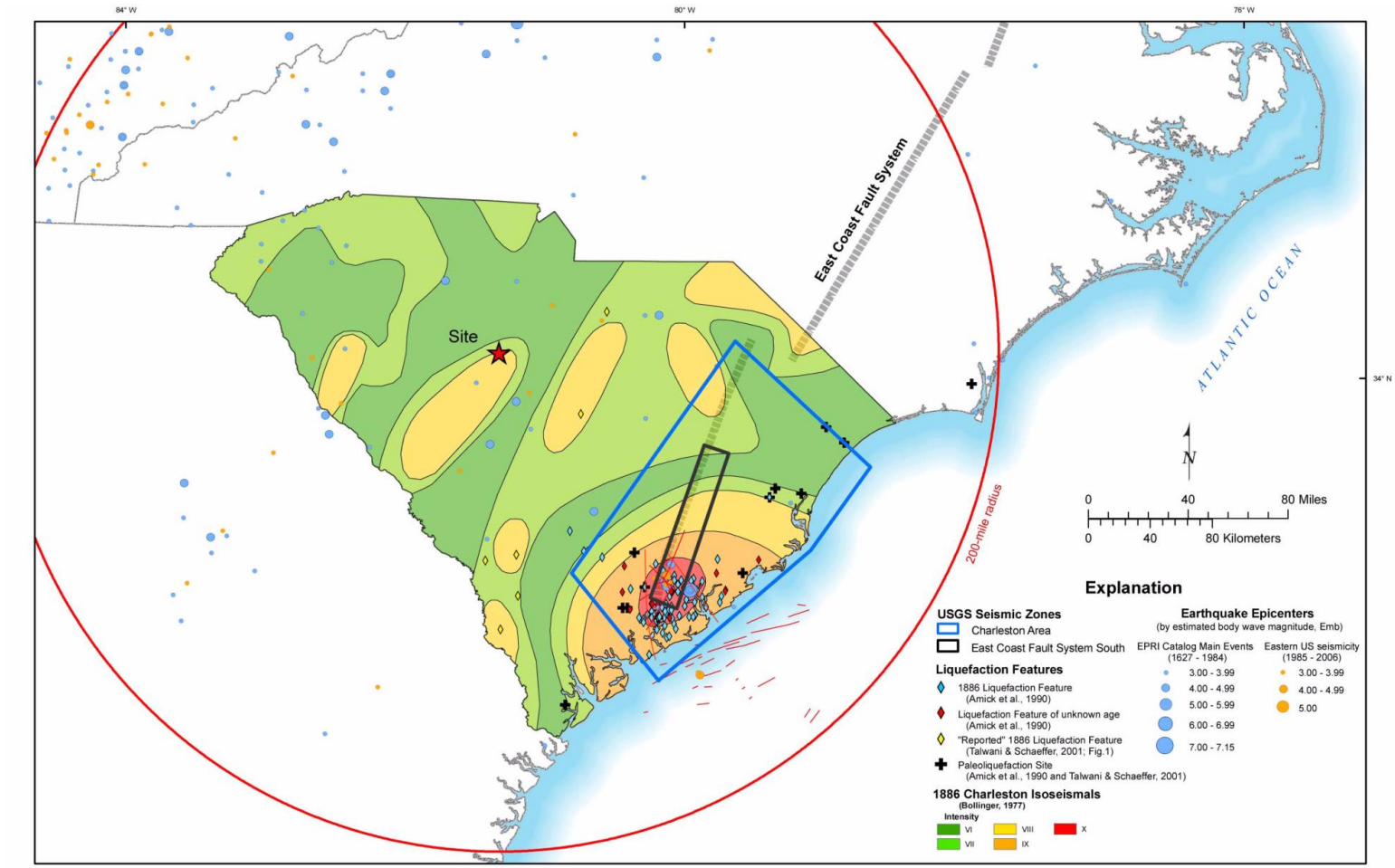
EPRI and USGS (2002) Seismic Source Model Comparison

- USGS M_{max} value higher than EPRI values on average for source zones outside Charleston (M=7.5 vs M=6.2*)
- Charleston seismic source models are similar
  - Maximum Magnitudes: M=7.2 (USGS) vs M=7.1* (EPRI)
  - Recurrence Interval: 550 yrs (USGS) vs 630 yrs* (EPRI)
  - Source Geometries

*average value from a distribution

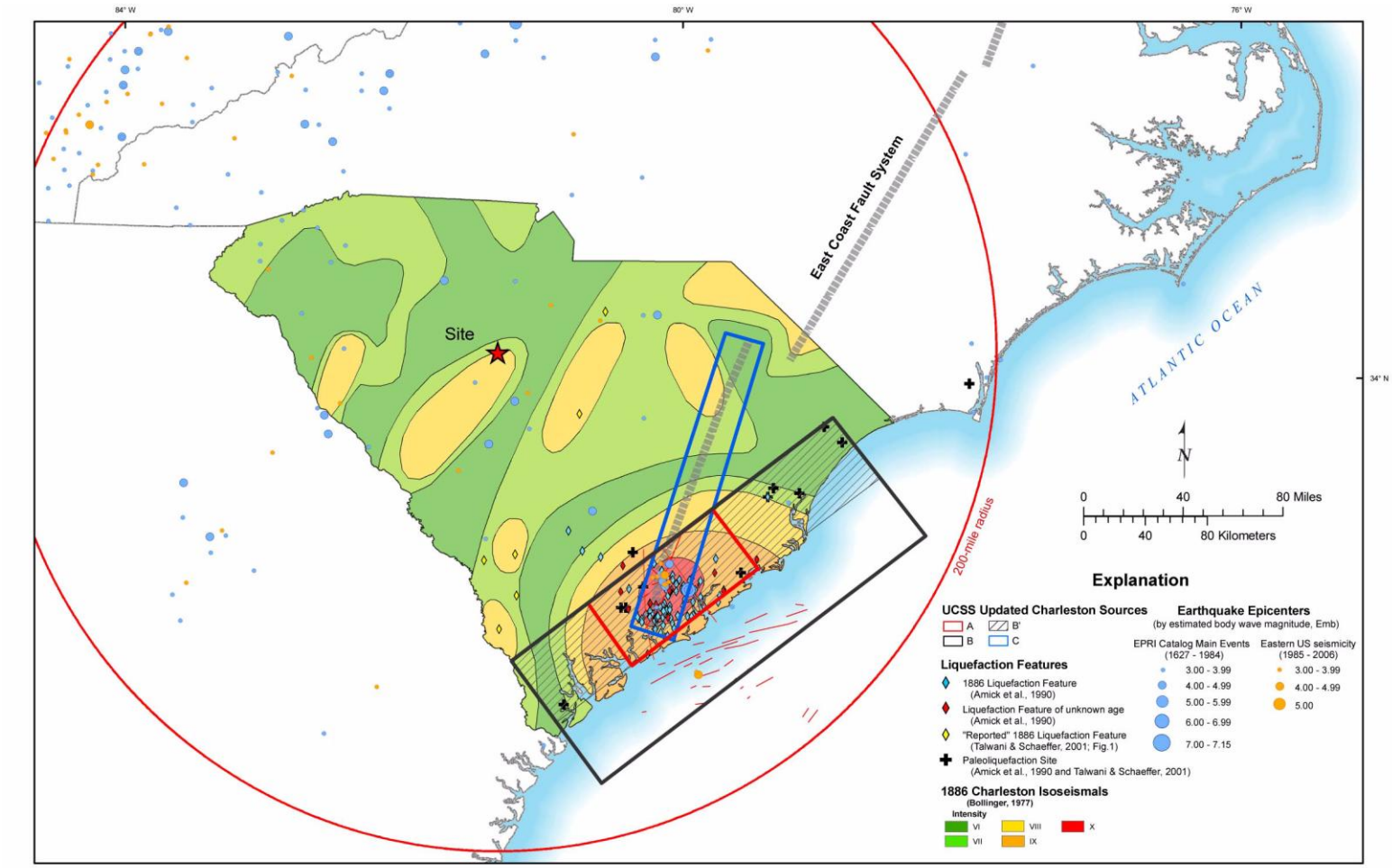
Attachment 3

USGS Source Model for Charleston



Attachment 3

Updated EPRI Source Model for Charleston



USGS (2008) Seismic Source Model

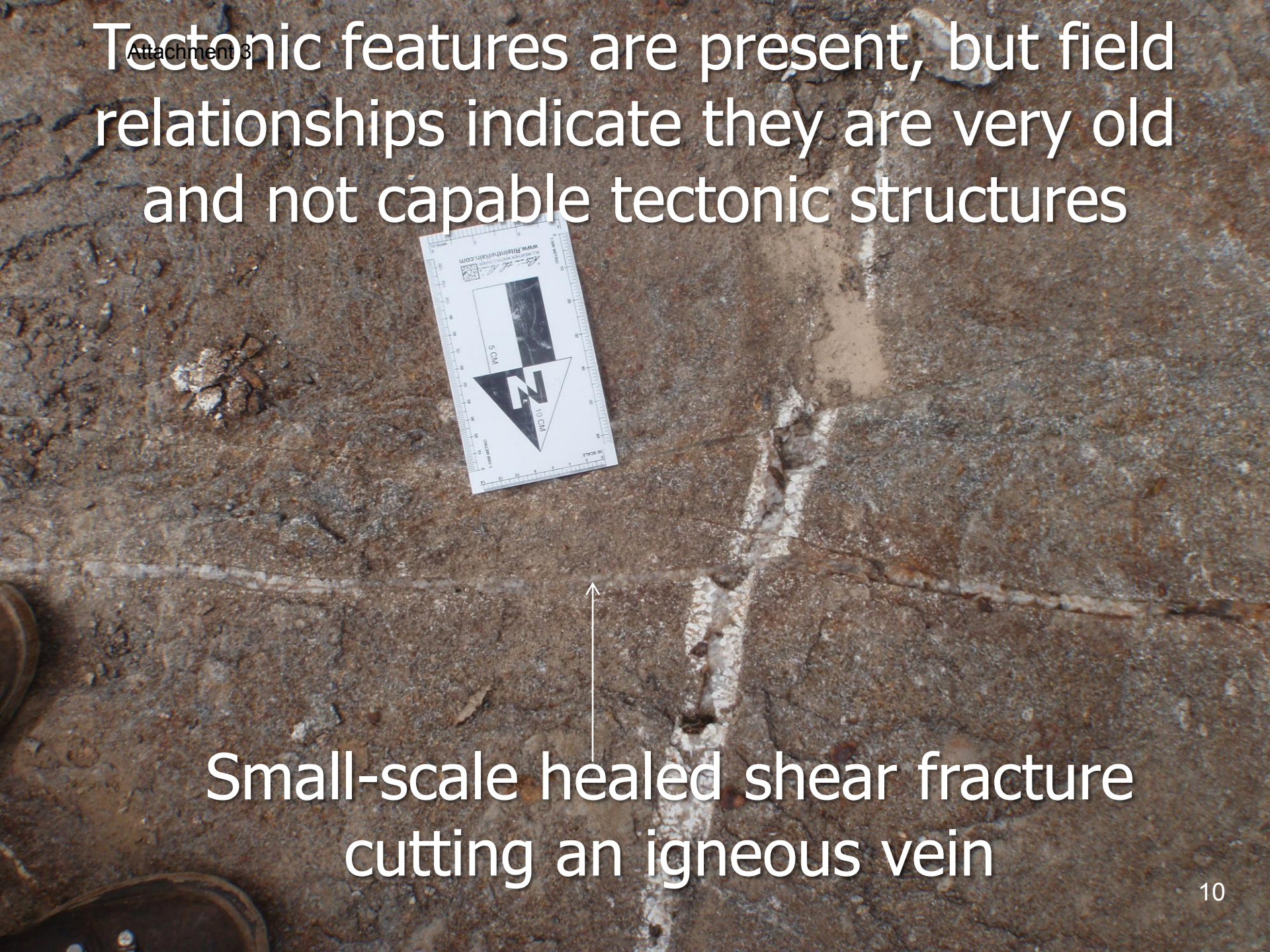
- Applicant compared EPRI seismic source model with USGS (2002) but not USGS (2008) models
- USGS (2008) updates
  - Maximum magnitude distribution replaced single values (M=7.5 vs M=7.1 to M=7.7)
  - Updated ground motion attenuation models
  - Charleston source model enlarged offshore
- Overall USGS (2008) results 10 to 15% lower than USGS (2002) for SE U.S. (USGS OFR 2008-1128)

2.5.1 Basic Geologic and Seismic Information

- **Update on observations by NRC geologists on geologic mapping of the Unit 2 excavation to assess the presence of tectonic features**
  - License Condition 2.5.1-1 requires the applicant to perform geologic mapping of excavations for safety-related structures; evaluate geologic features discovered; and notify NRC when excavations are ready for examination.
    - Minor shear zones proven by the applicant to be at least 45 Ma in age were mapped in the Unit 1 excavation, and similar features may occur in the excavations for Units 2 and 3.
  - In August 2010, staff directly examined geologic features being mapped by the applicant in the Unit 2 excavation to ensure that no capable tectonic structures existed therein.

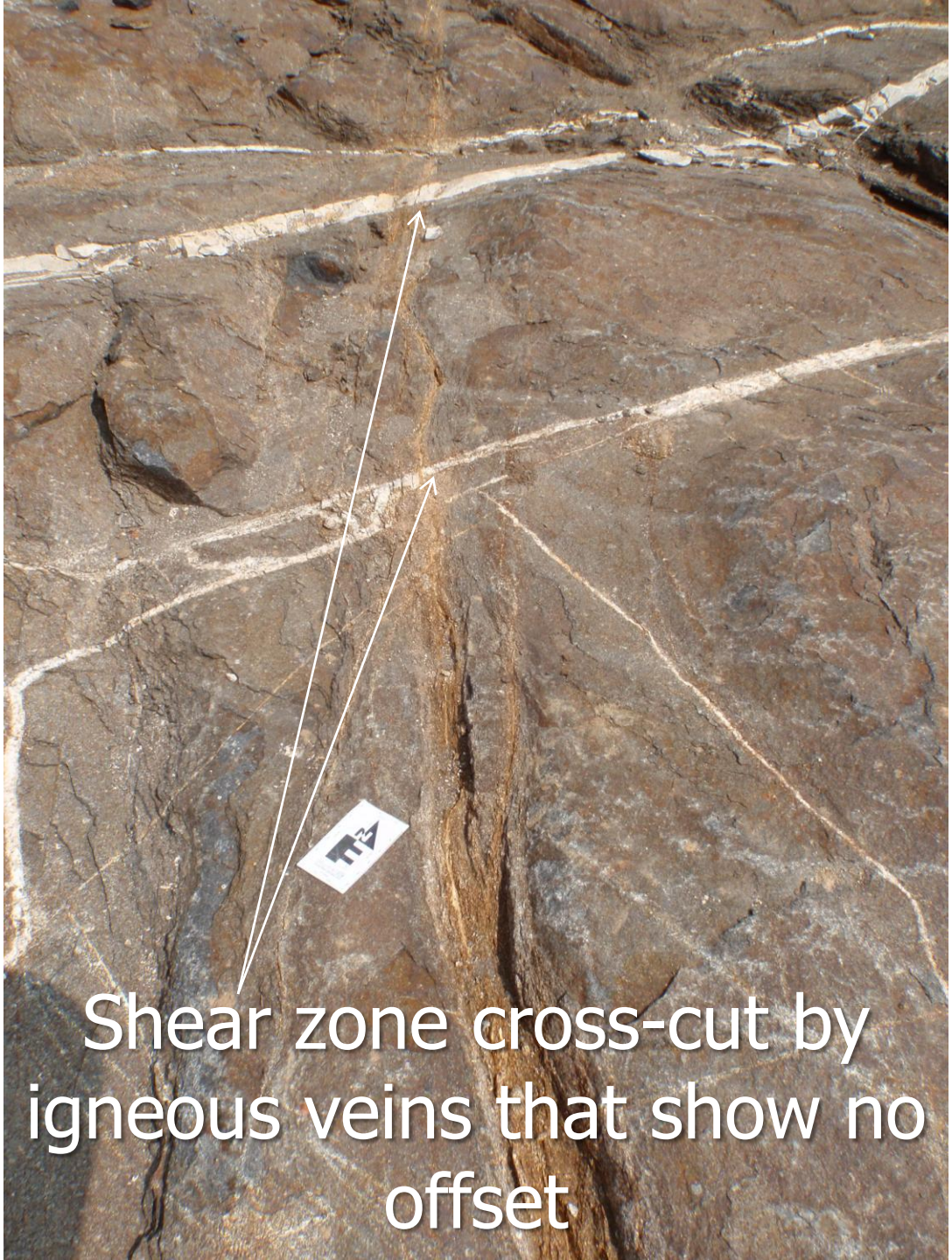
Potential tectonic features were carefully examined by NRC geologists

Tectonic features are present, but field relationships indicate they are very old and not capable tectonic structures



Small-scale healed shear fracture cutting an igneous vein

Attachment 3



2.5.1 Basic Geologic and Seismic Information

- **NRC geologists found that descriptions provided by the applicant in AFSAR Section 2.5 are fully consistent with geologic features observed in the Unit 2 excavation to date.**
  - A follow-up visit to the Unit 2 excavation by NRC geologists and a geotechnical engineer will occur after controlled blasting to reach the foundation level is completed.
  - Similar visits to carefully examine the Unit 3 excavation will also be conducted.

Attachment 3

VC Summer Units 2 and 3 FSAR Chapter 1

Introduction and Interfaces

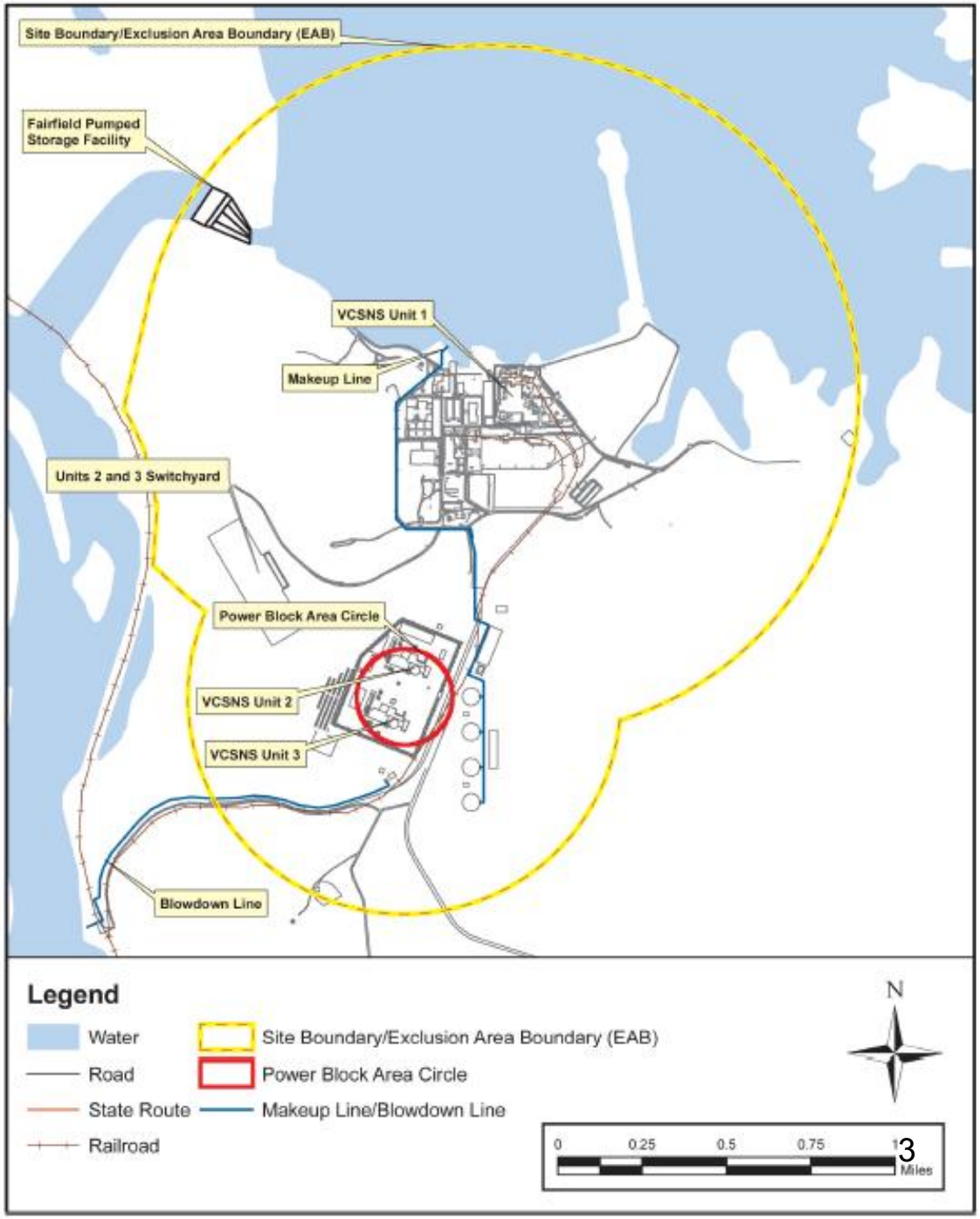
Amy M. Monroe – Licensing
New Nuclear Deployment

Chapter 1

- DCD Incorporated by Reference
- Standard material incorporated (including supplements, departures and exemptions)
- Additional site-specific material contained in Sections 1.2, 1.4, 1.8
- Discussion of departures and exemptions

Attachment 3

Section 1.2 Site Plan



Section 1.4 Identification of Agents and Contractors

- Co-owned with South Carolina Public Service Authority (Santee Cooper)
- EPC with Consortium – Westinghouse Electric Company and Shaw Group
- Other Technical Support

Section 1.8
Interfaces for Standard Design

- Departures -2 Standard and 3 VCSNS specific
- Exemptions – 2 Standard and 1 VCSNS specific

Questions?

Attachment 3

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

Summer Units 2 and 3 COL Application Review

Overview of Remaining Advanced Safety Evaluation (ASE) Report Chapter and ASE Chapter 1 Introduction and Interfaces

January 10 – 11, 2011

Overview of Remaining Chapters

- ACRS subcommittee meeting in July 2010
  - Discussed chapter 2 without hydrology
- Staff's philosophy for remaining presentations
  - The staff does not intend to brief the ACRS subcommittee on any standard content material.
  - Chapters that will not be presented include the following:
    - Chapter 4, "Reactor"
    - Chapter 7, "Instrumentation and Control"
    - Chapter 14, "Initial Test Program"

Overview of Remaining Chapters

- The staff does intend to provide a high-level description of the site-specific content on a chapter by chapter basis
  - The staff does not intend to brief every site-specific item, rather it intends to brief the ACRS on a subset of those issues, as appropriate.

Staff Review Team

- Technical Staff
  - Aaron Szabo, NRR
- Project Management
  - Joe Sebrosky, Projects

Summer COL Application

- Summer Application consists of:
  - Material incorporated by reference (IBR) from the AP1000 Design Control Document (DCD)
    - Staff's safety evaluation for the AP1000 design certification reflected in NUREG-1793 and its supplement
    - Staff's safety evaluation of AP1000 DC amendment was completed and presented to the committee
  - Standard content material (applicable to all AP1000 COL applicant)
    - Summer's safety evaluation for standard content references Vogtle's advanced safety evaluation report
      - Standard content evaluation material is double indented and italicized
      - Standard content evaluation contains some language from the Bellefonte safety evaluation report with open items to capture evaluations that were performed when Bellefonte was the reference COL
  - Summer plant specific information

Attachment 3

Summer COL Overview

Part Number	Description	Evaluation
1	General and Administration Information	Section 1.5.1
2	Final Safety analysis Report	In appropriate SER Chapters
3	Environmental Report	Final Environmental Impact statement
4	Technical Specifications	Chapter 16
5	Emergency Plan	Chapter 13
6	Limited Work Authorization	Not applicable
7	Departure Reports	In appropriate SER Chapters
8	Security Plan	Section 13.6
9	Withheld Information	In appropriate SER Chapters
10	Proposed Combined License Conditions (Including ITAAC)	In appropriate SER Chapters
11	Subsurface report detailing the results of geotechnical exploration	Section 2.5
12	Seismic Technical Advisory Group review letter	Section 2.5
13	Quality Assurance Program Description	Chapter 17
14	Mitigative Strategies Document for loss of large areas of the plant due to explosions or fire	Appendix 19.A
15	Cyber Security Plan	Section 13.8
16	Special Nuclear Material Control and Accounting Program Description	Section 1.5.5

Attachment 3

Overview of Summer COL FSAR Chapter 1

FSAR Section	Summary of Departures/Supplements
1.1 Introduction	Incorporated By Reference (IBR) with standard and site specific supplements
1.2 General Plant Description	IBR with site-specific supplements
1.3 Comparisons with Similar Facility designs	Completely IBR
1.4 Identification of Agents and Contactors	IBR with site-specific supplements
1.5 Requirements for Further Technical Information	Completely IBR
1.6 Material Referenced	IBR with standard and site-specific supplements
1.7 Drawings and Other Detailed Information	IBR with site-specific supplements
1.8 Interface for Standard Designs	IBR with site-specific supplements
1.9 Compliance with Regulatory Criteria	IBR with standard and site-specific supplements
1.10 Nuclear Power Plants to be Operated on Multi-Units Sites	Standard and site-specific supplemental information

Technical Topics of Interest

Summer COL Technical Topics of Interest

- Departures and Exemptions
  - **Departures**
    - COL application organization and numbering (Section 1.5.4)
    - COL application organization and numbering for FSAR chapter 2 (Section 2.0)
    - Departure for maximum wet bulb (noncoincident) air temperature (Sections 2.0, 2.3.1, 5.4, 6.2, 6.4, 9.1.3, 9.2.2 and 9.2.7)
    - Emergency response facility locations (Section 13.3)
    - Class 1E voltage regulating transformer current limiting features (Section 8.3.2)
  - **Exemptions**
    - COL application organization and numbering (Section 1.5.4 and 2.0)
    - Exemption from maximum safety wet bulb (noncoincident) air temperature (Section 9.2.2)
    - From requirements of 10 CFR 70.22(b), 70.32(c), and 10 CFR 74.31, 74.41 and 74.51(Section 1.5.4)

Technical Topics of Interest

Other Topics of Interest

- **Financial and Technical qualifications Review**
  - Technical qualification review in accordance with 10 CFR 52.97(a)(1)(iv) --- (Section 1.4.4)
  - Evaluates financial resources to build, operate and eventually decommission a nuclear facility in accordance with 10 CFR 52.79(a)(1)(iv)--(Section 1.5.1)

Attachment 3

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VC Summer Units 2 and 3 Chapters 5, 6 and 9

Site –Specific Wet Bulb Temperature Exemption

Amy M. Monroe – Licensing
New Nuclear Deployment
Mark Stella - Westinghouse

Basis for Exemption Request

- NRC RAI on site temperature limits generated during COLA review
- 100-year ambient wet bulb return temperature for site determined to exceed DCD maximum safety wet bulb limit
- Several areas potentially affected by the higher wet bulb temperature at the site

Wet-Bulb Temperature Exemption

- Site-specific maximum safety non-coincident wet-bulb temperature was determined to be 87.3ºF (1.2ºF above the AP1000 DCD Tier 1, Chapter 5, Table 5.0-1 value) based on the 100 year return value.

Evaluation of Impacts

- Evaluated AP1000 systems to determine those affected by change in maximum safety wet bulb temperature
- Assessed performance of systems and components affected by quantitative evaluations and calculations
- Performance of systems still acceptable with increased wet bulb temperature

AP1000 DCD Areas Potentially Affected and Outcomes of Assessments

- 6.2.2 – Passive Containment Cooling System Performance – *final pressure increase negligible compared to performance at standard plant limit*
- 5.4.7.1.2.3 – Normal Residual Heat Removal System – In-Containment Refueling Water Storage Tank temperature control - *final IRWST temperature after PRHR initiation increased by several degrees but remained well below boiling*

AP1000 DCD Areas Potentially Affected and Outcomes of Assessments

- 9.2.2.1.2.1 – Component Cooling System – Normal Operation temperature limit – *maximum CCS temperature increased by approximately 2.5 ^{o}F above nominal design temperature of 95 ^{o}F – remains below limiting temperature for acceptable RCP cooling*
- 9.2.2.1.2.2 – Component Cooling Water – Normal Plant Cooldown – *no impact as a result of increase in maximum safety wet bulb temperature*

AP1000 DCD Areas Potentially Affected (continued)

- 9.1.3.1.3.1 – Spent Fuel Pool Cooling –Partial Core shuffle (Normal refueling pool temperature control) – *slight increase but SFS pool temperature remains below 120 ^{o}F*
- 9.2.1.2.3.4 – Service Water System – Plant Cooldown/shutdown maximum cooling water temperature at peak heat load - *not affected by increase in maximum safety wet bulb temperature*

AP1000 DCD Areas Potentially Affected (continued)

- 9.2.2.1.2.3 – Component Cooling Water – Refueling (Full Core Offload) - *not affected by increase in maximum safety wet bulb temperature*
- 9.2.7.2.4 – Central Chilled Water System – Normal Operation - *effect of increased wet bulb temperature on MCR cooling, instrument and battery room cooling, and pump room cooling can be accommodated within the available capacity margin of the air-cooled chiller units*

Safety Systems Not Impacted

- Systems affected only by Maximum Safety Dry Bulb Temperature
- Systems whose performance is based on the Maximum Normal Non-coincident Wet Bulb Temperature or on the Coincident Maximum Dry Bulb and Wet Bulb Temperature

Questions?

Attachment 3

SCE&G • Santee Cooper
Shaw • Westinghouse Electric Company

VC Summer Units 2 and 3 FSAR Chapters 5, 6 and 9 Additional Information

Amy M. Monroe – Licensing
New Nuclear Deployment

Chapter 5
Reactor Coolant System and Connected Systems

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific Wet Bulb exemption discussed previously - no additional non-standard information contained in FSAR

Chapter 6
Engineered Safety Features

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific Wet Bulb exemption (discussed previously)
- All chemical hazard evaluations are discussed in FSAR 2.2.3

Chapter 9
Auxiliary Systems

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific Wet Bulb exemption (discussed previously)
- Site Specific Systems of Interest

Chapter 9
Auxiliary Systems

- Service Water System Cooling Towers
- Raw Water System has no safety related function and failure of the system will not affect the ability of a safety system to perform its function.

Questions?

Attachment 3

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

V. C. Summer Units 2 and 3
COL Application Review

Departure and Exemption for Wet Bulb Non-coincident Temperature

ASE Chapters 5, 6, and 9

Reactor Coolant System and Connected Systems,
Engineered Safety Features, and
Auxiliary Systems

January 10 -11, 2011

Staff Review Team

- Technical Staff
  - Chapter 5, Steam and Power Conversion (Reactor Systems, Nuclear Performance and Code Review Branch)
    - John Budzynski
  - Chapter 6, Engineered Safety Features (Containment & Ventilation Branch)
    - Michelle Hayes
    - Shie-Jeng Peng
  - Chapter 9, Auxiliary Systems (Balance of Plant Branch)
    - Larry Wheeler
    - Raul Hernandez
- Project Managers
  - Joe Sebrosky, Don Habib, Sujata Goetz

Overview of Wet Bulb Departure and Exemption

- Evaluations Affected
  - 5.4.7, Normal residual heat removal system
  - 6.2, Containment systems
  - 6.4, Habitability systems (for main control room)
    - Nuclear island nonradioactive ventilation system (VBS)
    - Low capacity chilled water system (LCCWS)
  - 9.1.3, Spent fuel pool cooling system (SFS) - nonsafety
  - 9.2.2, Component cooling water system (CCS) - RTNSS
  - 9.2.7, Central Chilled Water system (VWS) - nonsafety

- COL Revision 2, maximum safety wet-bulb (noncoincident) air temperature increased from 86.1°F to 87.3°F
  - Based on 100 year return temperature (Chapter 2)

- Maximum coincident wet bulb temperature (86.1°F) and maximum dry bulb temperature (115°F) have not changed from the standard AP1000 values

Section 5.4.7, Normal Heat Removal System (RNS)

- Concern: Impact on the design basis
  - Plant cool-down from 350ºF to 125ºF in 96 hours
  - IRWST temperature
    - <120ºF (normal operation)
    - <boiling (during extended operation)

- The NRC staff concluded that the proposed change in the maximum safety non-coincident wet bulb temperature does not impact the residual heat removal system (RNS) capacity to perform its functions as described in DCD Section 5.4.7.

Section 6.2, Containment Systems

- Will the increased wet-bulb temperature affect the performance of the containment system?
- WGOTHIC analysis demonstrated no impact to peak containment pressure reported in DCD
- Staff CONTAIN analysis confirmed results. Also confirmed no change to air only cooling case.

Section 6.4, Habitability Systems

- **Issue:**
  - Will the higher wet bulb temperature have safety-significant impact on the control room habitability (SRP Sec. 6.4)?

- **Concern:**
  - Maximum safety temperatures ➔ LCCWS ➔ VBS ➔ MCR HVAC
  - SRP 6.4: GDC 4, “Environmental and Dynamic Effects Design Bases”; GDC 19, “Control Room”

Section 6.4, Habitability Systems

- **RAI Response:**
  - Bounded calculation (87.4ºF wet bulb temperature)
  - Design margin

- **Review:**
  - Audit calculation note and conduct public meeting
  - Staff finds that the applicant has provided reasonable assurance that the increase of wet bulb temperature of 1.2ºF would not have safety-significant impact on the control room habitability. Staff concludes from control room habitability aspect that the departure is acceptable.

SFS, CCS, and VWS (Chapter 9, Auxiliary Systems)

- Normal CCS temperature <95ºF with wet bulb 84ºF or lower; as wet bulb temperature increases, CCS temperature increases
- Calculations reviewed by staff at audit
- Same methodology used for Westinghouse TR-36 (wet bulb changes to support AP1000 DCD Revision 16)
- No equipment changes were needed

Section 9.1.3, Spent Fuel Pool Cooling System

- Design parameter SFP < 120°F (AP1000 DCD 9.1.3)
- CCS water temperature rises from 97°F to 97.3°F (Δ0.3°F) due to increase in wet bulb to 87.3°F
- CCS water temperature of 97.3°F, spent fuel pool (SFP) temperature remains below 115°F
- Staff concludes SFP remains within design parameter of <120°F

Section 9.2.2, Component Cooling Water System

- AP1000 DCD Section 9.2.2.1.2.1, normal CCS supply temperature to plant components is not more than 100°F
- Normal CCS water temperature <95°F with wet bulb at 84°F or lower; as wet bulb temperature increases, CCS temperature increases
- CCS water temperature rises from 97°F to 97.3°F (Δ0.3°F) due to increase in wet bulb to 87.3°F
- Higher wet bulb temperature conditions are expected to be of short duration ; periods of <2 hours (estimated to occur 30 hours per year)

Section 9.2.2, Component Cooling Water System (continued)

- Reactor coolant pump motors limited to 100°F for 6 hours
- CCS RTNSS function Mode 5/6 to remove decay heat, significant lower heat loads and no RCPs operating
- Reactor cooling system cooldown uses 80.1°F wet bulb for CCS
- Staff concludes CCS remains within design parameter of < 100°F

Section 9.2.7, Central Chilled Water System (VWS)

- VWS supplies chilled water to various HVAC systems (nonsafety system)
- Two closed loop subsystem
  - High capacity chilled water (HCCWS)
    - Majority of plant HVAC system
      - Radwaste building, control access area, auxiliary building, etc.
  - Low capacity chilled water (LCCWS)
    - Supplies nuclear island nonradioactive ventilation system (VBS)
      - Main Control Room
      - Chemical and volume control system makeup pumps
      - Normal residual heat removal pump compartments unit coolers

Section 9.2.7, Central Chilled Water System (VWS) (continued)

- The VBS is the only HVAC system designed to accommodate the maximum safety temperature limits
- Higher maximum safety ambient wet bulb temperature of 87.3°F can be accommodated within the available capacity margin of the chiller units
- HVAC calculations reviewed by staff at audit, 164 ton load revised to 182 tons; equipment rated at 300 tons – no equipment changes required
- VBS air handling unit has cooling coil and system margin
- Staff concludes VBS has adequate system margins

Staff Conclusions for SFS, CCS, and VWS

- Increasing maximum safety wet-bulb (noncoincident) air temperature from 86.1°F to 87.3°F is acceptable since:
- SFP < 120°F (AP1000 design parameter)
- CCS < 100°F (AP1000 design parameter)
- Existing VBS margin remains adequate

Wet Bulb Exemption

- Exemption requested from 10 CFR Part 52, Appendix D, Section IV.A.2.d related to demonstrating compliance with site parameters

- In order to grant the exemption, the staff determined the following:
  - The exemption does not have an adverse impact on the AP1000 standard plant design and therefore will not result in a significant decrease in safety otherwise provided by the design
  - The exemption is not inconsistent with the Atomic Energy Act or any other statue and therefore is authorized by law
  - Special circumstances are present as specified in 10 CFR 50.12(a)(2).
    - Staff found that special circumstance 10 CFR 50.12(a)(2)(ii) applied, (i.e., application of the regulation is not needed to achieve the underlying purpose of the rule)
  - The special circumstances outweigh any decrease in safety that may result from the reduction in standardization

Summer FSAR Chapter 5

Reactor Coolant System and Connected Systems

FSAR Section	Site-Specific Evaluations
5.2.1.1 Compliance with 10 CFR 50.55a 5.2.1.2 Applicable Code Cases 5.2.1.3 Alternate Classification 5.2.2 Overpressure Protection 5.2.3 Reactor Coolant Pressure Boundary Materials 5.2.4 Inservice Inspection and Testing of Class 1 Components 5.2.5 Detection of Leakage through Reactor Coolant Pressure Boundary 5.3.1 Reactor Vessel Design 5.3.2 Reactor Vessel Materials 5.3.3 Pressure Temperature Limits 5.3.4 Reactor Vessel Integrity 5.3.5 Reactor Vessel Insulation	• None*
5.4 Component and Subsystem Design	• VCS DEP 2.0-2, Maximum Safety Wet Bulb (Noncoincident) Air Temperature

* This section is entirely IBR or IBR/standard.

Summer FSAR Chapter 6

Engineered Safety Features

FSAR Section	Site-Specific Evaluations
6.1.1 Engineered Safety Materials Features, Metallic Materials	• None *
6.1.2 Engineered Safety Materials Features, Organic Materials	• None *
6.2 Containment Systems	• VCS DEP 2.0-2, Maximum Safety Wet Bulb (Noncoincident) Air Temperature
6.3 Passive Core Cooling System	• None *
6.4 Habitability Systems	• ACRS Action Item #63, Staff confirmatory calculation regarding VCS COL 6.4-1, Concentrations of Site-Specific Chemicals • VCS DEP 2.0-2, Maximum Safety Wet Bulb (Noncoincident) Air Temperature
6.5 Fission Product Removal and Control Systems	• None *
6.6 Inservice Inspection of Class 2, 3, and MC Components	• None *

* This section is entirely IBR or IBR/standard.

Summer FSAR Chapter 9

Auxiliary Systems

FSAR Section	Site-Specific Evaluations
9.1.1 New Fuel Storage	• None*
9.1.2 Spent Fuel Storage	• None*
9.1.3 Spent Fuel Pool Cooling System	• VCS DEP 2.0-2, Maximum Safety Wet Bulb (Noncoincident) Air Temperature
9.1.4 Light Load Handling System	• None*
9.1.5 Overhead Heavy Load Handling Systems	• None*
9.2.1 Service Water System	• VCS SUP 9.2-3 provides additional information regarding the service water system cooling tower potential interactions
9.2.2 Component Cooling Water System	• VCS DEP 2.0-2, Maximum Safety Wet Bulb (Noncoincident) Air Temperature
9.2.3 Demineralized Water Treatment System	• None*
9.2.4 Demineralized Water Transfer and Storage System	• None*

* This section is entirely IBR or IBR/standard.

Summer FSAR Chapter 9

Auxiliary Systems

FSAR Section	Site-Specific Evaluations
9.2.5 Potable Water System	• VCS COL 9.2-1, Potable water system description outside the power block
9.2.6 Sanitary Drains	• VCS SUP 9.2-1, Sanitary waste system discharge description
9.2.7 Central Chilled Water System	• VCS DEP 2.0-2, Maximum Safety Wet Bulb (Noncoincident) Air Temperature
9.2.8 Turbine Building Closed Cooling Water System (TCS)	• VCS CDI provides the source of cooling water for the TCS heat exchangers
9.2.9 Waste Water System	• VCS COL 9.2-2 provides information on the waste water retention basins and associated discharge piping
9.2.10 Hot Water Heating System	• None*
9.2.11 Raw Water System	• VCS SUP 9.2-2 provides site-specific information related to the raw water system
9.3.1 Compressed and Instrument Air System	• None*
9.3.2 Plant Gas System	• None*

* This section is entirely IBR or IBR/standard.

Summer FSAR Chapter 9

Auxiliary Systems

FSAR Section	Site-Specific Evaluations
9.3.3 Primary Sampling System	• None*
9.3.4 Secondary Sampling System	• None*
9.3.5 Equipment and Floor Drainage Systems	• None*
9.3.6 Chemical and Volume Control System	• None*
9.4.1 Nuclear Island Nonradioactive Ventilation System	• VCS COL 9.4-1b provides local toxic gas evaluations
9.4.2 Annex/Auxiliary Buildings Nonradioactive HVAC System	• None*
9.4.6 Containment Recirculation Cooling System	• None*
9.4.7 Containment Air Filtration System	• None*
9.4.8 Radwaste Building HVAC System	• None*

* This section is entirely IBR or IBR/standard.

Summer FSAR Chapter 9

Auxiliary Systems

FSAR Section	Site-Specific Evaluations
9.4.9 Turbine Building Ventilation System	• None*
9.4.10 Diesel Geneartor Building Heating and Ventilation System	• None*
9.4.11 Health Physics and Hot Machine Shop HVAC System	• None*
9.5.1 Fire Protection System	• VCS COL 9.5-1, qualification requirements for the fire protection program • VCS COL 9.5-2, site-specific hazards analysis of the yard areas and outlying buildings
9.5.2 Communication System	• VCS COL 9.5-9, offsite interfaces • VCS COL 9.5-10, emergency offsite communications • VCD COL 9.5-11, security communications
9.5.3–9.58	• None*

* This section is entirely IBR or IBR/standard.

RWS Description

- **RWS is non-safety and non-RTNSS**

- **Raw water intake structure includes 3 non-safety pumps which pumps unfiltered water from the Monticello Reservoir to:**
  - CWS cooling towers
  - Alternate water for the SWS cooling towers via cross connect from water treatment header

- **Nearby offsite water treatment facility provides filtered water to:**
  - Demineralized water treatment system
  - Fire protection
  - Normal makeup to SWS cooling towers
  - Other misc users such as condenser vacuum pump seal water heat exchanger and TBCCW heat exchanger cooling

RWS Description

- **RWS is a shared system for Unit 2 & 3 which includes:**
  - Offsite water treatment ~ 1000 gpm
  - 400,000 acre-feet of reservoir (adequate to support 7 days of CDS operations)
  - 3 -50% capacity raw water pumps to support normal CWS makeup (alternate to SWS)
  - 2- 100% capacity screen wash pumps
  - 2 of the 3 raw water pumps and discharge valves are diesel backed
  - Traveling screens and screen wash pumps are diesel backed
  - HDPE underground piping materials

Staff Review Summary

- RWS has redundancy with RWS pumps to support CSD
- Reliable materials are being utilized consistent with industry good practices
- RWS is non radioactive and contamination is not credible due to its configuration relative to potential sources of contamination

Staff Review Summary

- **GDC 2 and GDC 4 have been satisfied**
  - Failure of the RWS/components will not affect the ability of any risk-significant systems to perform their intended safety functions
  - Failure of the RWS/components will not affect any RTNSS
- **Staff concludes that RWS:**
  - Meets all applicable regulations
  - Considered highly reliable to support CSD

Attachment 3

VC Summer Units 2 and 3 FSAR Chapters 3 and 19

Amy M. Monroe – Licensing
New Nuclear Deployment

Chapter 3
Design of Structures, Components, Equipment and Systems

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific supplements
  - 3.3 - Wind and Tornado Loadings
  - 3.4 – Flooding
  - 3.5 - Turbine Missiles
  - 3.7 – Seismic Design
  - 3.8 – Waterproofing Material

Chapter 19
Probabilistic Risk Assessment

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific external events evaluation

Section 19.58
External Events

- Winds
- Floods
- Transportation and Nearby Facility Accidents
- Fires

Questions?

Attachment 3

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

Virgil C. Summer Units 2 and 3 COL Application Review

ASE Chapters 3 and 19

Design of Structures, Components, Equipment and Systems, and Probabilistic Risk Assessment

January 10–11, 2011

Staff Review Team

- **Technical Staff**

  Chapter 3

  – Bret Tegeler, Structural Engineering Branch

  – Vaughn Thomas, Structural Engineering Branch

  Chapter 19

  – Malcolm Patterson, PRA and Severe Accidents Branch

- **Project Managers**

  – Tom Galletta, chapter 19

  – Terri Spicher, chapter 3

Summer FSAR Chapter 3

Design of Structures, Components, Equipment and Systems

FSAR Section	Site-Specific Evaluations
3.1 Conformance With Nuclear Regulatory Commission General Design Criteria	• None*
3.2 Classification of Structures, Components, and Systems	• None*
3.3 Wind and Tornado Loadings	• VCS COL 3.3-1 Wind Velocity Characteristics • VCS COL 3.5-1 Tornado Velocity Characteristics
3.4 Water Level (Flood) Design	• VCS COL 3.4-1 Dewatering System and Water Levels
3.5 Missile Protection	• VCS SUP 3.5-1 Turbine Missile from Unit 1

* This section is entirely IBR or IBR/standard

Summer FSAR Chapter 3

Design of Structures, Components, Equipment and Systems

FSAR Section	Site-Specific Evaluations
3.6 Protection Against the Dynamic Effects Associated With the Postulated Rupture of Piping	• None*
3.7 Seismic Design	• VCS SUP 3.7-3 Design Ground Motion Response Spectra
	• VCS COL 3.7-1 Seismic Analysis of Dams
3.8 Design of Category I Structures	• VCS COL 2.5-17 Waterproofing Material for Category I Structures
3.9 Mechanical Systems and Components	• None*
3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment	• None*
3.11 Environmental Qualification of Mechanical and Electrical Equipment	• None*

* This section is entirely IBR or IBR/standard.

Summer FSAR Chapter 19

Probabilistic Risk Assessment

FSAR Section	Site-Specific Evaluations
19.1–19.54, 19.56, 19.57	• None*
19.55 Seismic Margins Analysis	• VCS SUP 19.59.10-6 Site-Specific Seismic Margin Analysis
19.58 Winds, Floods, and Other External Events	• VCS SUP 19.58-1 External Event Frequencies
19.59 PRA Results and Insights	• None*

* This section is entirely IBR or IBR/Standard

VCS SUP 3.7-3 and SUP 19.59.10-6 Design Ground Motion Response Spectra

- Issue
  - COL applicant to provide a comparison of the site-specific ground motion response spectra (GMRS) to the hard rock high frequency (HRHF) spectra and Certified Seismic Design Response Spectra (CSDRS).
  - Site specific horizontal and vertical GMRS exceeds the standard AP1000 CSDRS at high frequencies; however, it is entirely bounded by the AP1000 HRHF spectra.
- Resolution
  - The staff concludes the high frequency seismic input was evaluated in the AP1000 DCD and considered to be non-damaging.

VCSNS GMRS vs. CSDRS

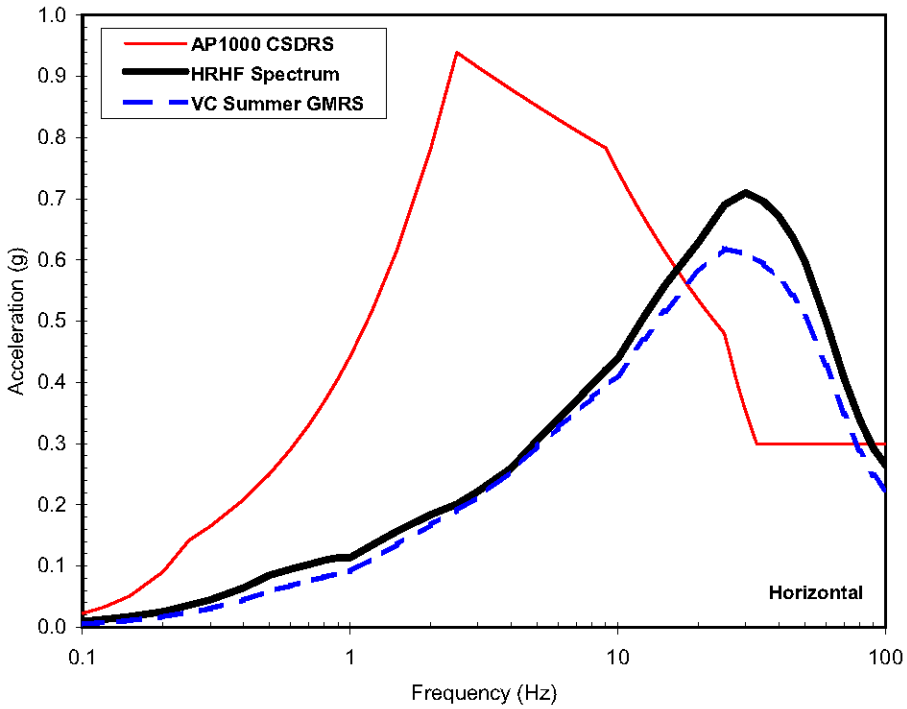


Figure 2.0-201. Comparison Plot of V. C. Summer GMRS and HRHF Spectra for the Horizontal Component of Motion

Technical Topics of Interest for VCS

- Issue
  - COL applicant to provide a summary of external events to confirm the basis for concluding that the VCS site was bounded by the generic AP1000 analysis.
- Resolution
  - Based on the parameters of the VCS site, provided in a plant-specific supplement, the staff confirmed that all external events that should be assessed may be screened from further evaluation. The staff concludes that the incorporation of AP1000 DCD Section 19.58 by reference is acceptable.

V.C. Summer External Events

External Event	Screening Criteria Applied			
	Bounded	Negligible Frequency	Negligible Consequence	Not Applicable
Tornado	•			
Hurricane	•		•[1]	
External flood				PMP flood < 100’ (grade)
Aviation		•		
Marine				No barge traffic
Pipeline	•			Nearest pipeline >1 mi.
Railroad			•	$D_{closest\ track} > D_{standoff}$
Truck	•			
Nearby facilities			•[2]	
External fires			•[2]	

[1] Extratropical cyclones
[2] Confirmatory items

Attachment 3

SCE&G • Santee Cooper
Shaw • Westinghouse Electric Company

VC Summer Units 2 and 3 FSAR Chapters 8 and 10

James LaBorde – Consulting Engineer
New Nuclear Deployment
Amy M. Monroe – Licensing
New Nuclear Deployment

Chapter 8
Electric Systems

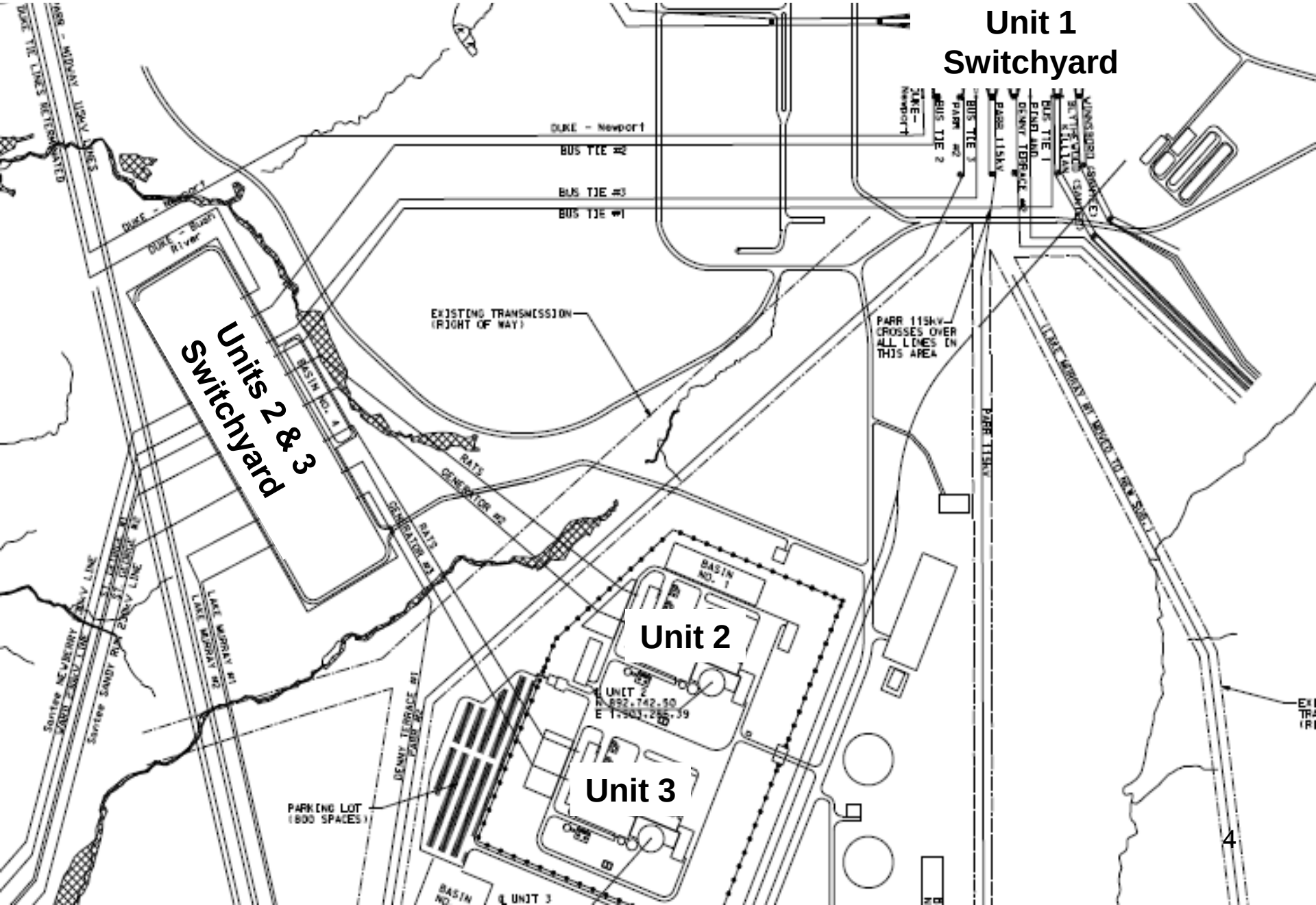
- DCD Incorporated by Reference
- Standard material incorporated (including standard departure)
- Site-specific off-site power description

Section 8.2
Offsite Power

- 12 overhead transmission lines connect the new 230 kv switchyard to other substations
- Switchyard is robust
- Failure Analysis performed
- Grid Stability Study performed
  - Includes the Westinghouse interface requirement for maintaining Reactor Coolant Pump voltage for 3 seconds after a turbine trip

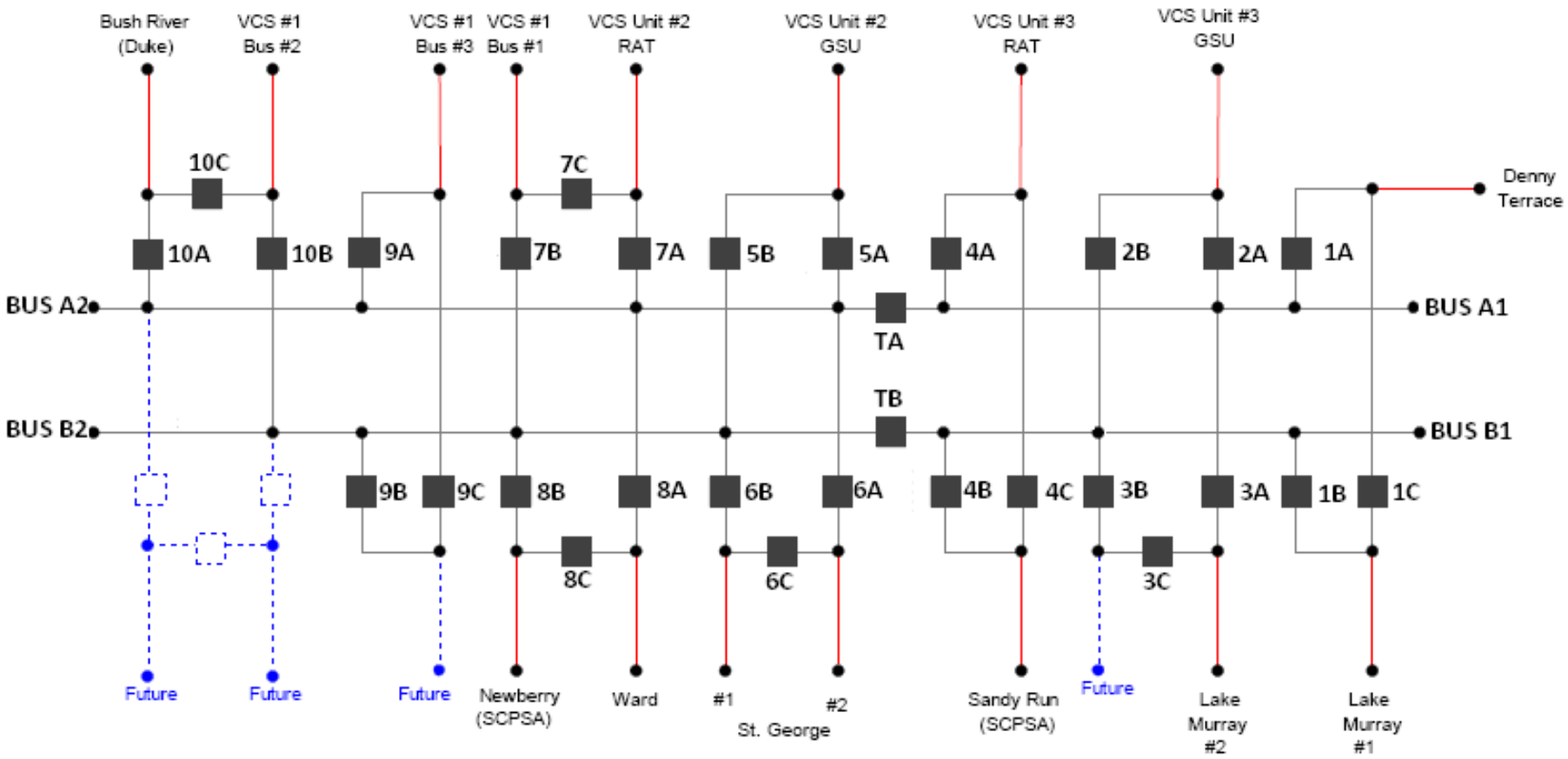
Attachment 1

Units 1, 2, & 3 Transmission Lines



Attachment 3

Switchyard Single-line Diagram



Questions?

Chapter 10
Steam and Power Conversion

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific discussions of interest
  - Circulating Water System (CWS) – Serves no safety-related function but is a heat sink for waste heat from the turbine discharge to the main condenser

Attachment 3

Circulating Water System Pipe

Facts:

10’ Diameter, 16’ Length, Weighs 64,000lbs

Questions?

Attachment 3

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

V. C. Summer Units 2 and 3 COL Application

ASE Chapters 8 and 10

Electric Power

Steam and Power Conversion

January 10-11, 2010

Staff Review Team

- Technical Staff
  - **Tania Martinez Navedo,** Electrical Engineer
  - **Om Chopra,** Electrical Engineer
- Project Manager
  - Joe Sebrosky

Summer FSAR Chapter 8

Electric Power

FSAR Section	Site-Specific Evaluations
8.1 Introduction	• VCS SUP 8.1-1 Summer Units 2 and 3 connection to the utility grid • VCS SUP 8.1-2 Additional information on regulatory guidelines and standards
8.2 Offsite Power System	• VCS COL 8.2-1 Transmission system description, and its testing and inspection plan • VCS COL 8.2-2 Switchyard description and protection relaying • VCS SUP 8.2-1 FMEA of the switchyard • VCS SUP 8.2-2 Transmission system requirements and studies • VCS SUP 8.2-3 Transmission system planning • VCS SUP 8.2-4 Stability and reliability of the offsite transmission power system • Interface Requirements
	• VCS Conceptual Design Information (CDI) describing the transformer area located next to each unit's turbine building

Summer FSAR Chapter 8

Electric Power

FSAR Section	Site-Specific Evaluations
8.3.1 AC Power Systems (Onsite)	• VCS COL 8.3-1 Grounding system and lightning protection
	• VCS SUP 8.3-1 Site-specific switchyard and power transformer voltage • VCS SUP 8.3-2 EDG rating based on site conditions
8.3.2 DC Power Systems (Onsite)	• None*

*This section is entirely IBR or IBR/Standard

Staff Review Summary

- **Section 8.1 – Introduction**
  - Applicant has adequately addressed VCS SUP 8.1-1 regarding V.C. Summer 2 and 3 Units' connection to the South Carolina Electric and Gas transmission system.
  - The applicant has adequately addressed VCS SUP 8.1-2 regarding additional information for regulatory guidelines and standards.

Staff Review Summary

- **Section 8.2 – Offsite Power System**
  - The staff finds COL information items VCS COL 8.2-1 involving the design details of the plant site switchyard and its interface with the local transmission grid adequately addressed pending closure of Confirmatory Item 8.2-1.
    - Confirmatory Item 8.2-1 relates to FSAR changes addressing interface items
  - The staff concludes that the applicant's condition monitoring program for underground or inaccessible cables satisfies the recommendations of GL 2007-01,and the guidance in NUREG/CR-7000 and NUREG-0800 pending closure of Confirmatory Item 8.2-3.

Staff Review Summary

- **Section 8.2 – Offsite Power System**
  - The applicant has adequately addressed VCS SUP 8.2-1 thru 8.2-4 involving the offsite power system adequacy and availability, testing and inspection of switchyard components and failure modes and effects analysis.
  - The applicant provided sufficient information regarding the interfaces for standard design from the generic AP1000 DCD, Table 1.8-1, Items 8.1, 8.2, and 8.3.

Staff Review Summary

- **Section 8.3.1 – AC Power System (Onsite)**
  - The applicant has adequately addressed VCS SUP 8.3-1 involving the site-specific switchyard and transformer voltage.
  - The applicant has adequately addressed VCS SUP 8.3-2 involving the site-specific conditions bounded by the standard site conditions in the AP1000 DCD for rating the diesel generator.

Summer FSAR Chapter 10

Steam and Power Conversion

FSAR Section	Site-Specific Evaluations
10.1 Summary Description	• None*
10.2 Turbine Generator	• None*
10.3 Main Steam Supply System	• None*
10.4 Other Features of Steam and Power Conversion System	• VCS CDI, relating to COL Section 10.4.2 for the site specific cooling water source for the vacuum pump seal water heat exchangers. • VCS CDI, relating COL Section 10.4.5 for the site specific Circulating Water System design information. • VCS COL 10.4-1 relating to the Circulating Water System design parameters. • VCS COL 10.4-2 relating to Condensate, Feedwater and Auxiliary Steam System Chemistry Control.

* This section is entirely IBR or IBR/standard.

Attachment 3

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VC Summer Units 2 and 3 FSAR Chapters 11, 12 and 13 (without Emergency Planning)

Amy M. Monroe – Licensing
New Nuclear Deployment

Chapter 11
Radioactive Waste Management

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific information of interest
  - Liquid Radwaste System (WLS) and Waste Water System (WWS)
  - Gaseous Waste Management
  - Effluent Monitoring

Section 11.2
Waste Management

- WLS connects to WWS within the Exclusion Area Boundary for dilution to ensure release limits are met.
- LADTAP II code utilized to determine estimated dose rates and doses.
  - Doses to individuals due to liquid waste discharges are within the limits of 10 CFR Part 50 Appendix I
  - Based on estimated population doses a liquid cost benefit analysis was performed and no augments were determined to be cost beneficial.

Section 11.3
Gaseous Waste Management

- GASPAR II computer code used to calculate dose and dose rates.
  - Gaseous waste discharge doses are within the regulatory limits of 10 CFR Part 50 Appendix I
  - Based on estimated population doses a gaseous radwaste cost benefit analysis was performed and no augments were determined to be cost beneficial.

Section 11.5
Radiation Monitoring

- SCE&G is extending the VCSNS Unit 1 quality assurance of radiological effluent and environmental monitoring program based on RG 4.15 Revision 1 to Units 2 and 3.

Chapter 12
Radiation Protection

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific departures to address TSC and OSC relocations (Figure changes)
- Site –specific evaluations of dose to construction workers

Section 12.4
Dose Assessment

- Doses to construction workers
  - Direct radiation as well as liquid and gaseous radioactive effluents from Unit 1 on Units 2 and 3 workers
  - Due to construction overlap, direct radiation as well as liquid and gaseous radioactive effluents from Unit 2 on Unit 3 workers

Chapter 13 Excluding 13.3 Conduct of Operations

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific issues
  - Organizational Structure

Questions?

Attachment 3

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

Presentation to the ACRS Subcommittee

V. C. Summer Units 2 and 3 COL Application Review

Chapters 11,12, and 13

Radioactive Waste Management,
Radioactive Protection, and
Conduct of Operations

January 10 -11, 2011

Attachment 3

Summer FSAR Chapter 11
Radioactive Waste Management

FSAR Section	Site-Specific Evaluations
11.1 Source Term	• None*
11.2 Liquid Radioactive Waste Management	• VCS COL 11.2-2, Liquid waste discharge cost-benefit analysis • VCS COL 2.4-5 and VCS 15.7-1, Doses from accidental release from liquid waste tank failure • VCS COL 11.5-3, Compliance with 10 CFR Part 50, Appendix I, Sections II.A and II.D for liquid waste discharges • VCS SUP 11.2-1, Liquid waste discharge pipe
11.3 Gaseous Radioactive Waste Management	• VCS COL 11.3-1, Gaseous waste discharge cost-benefit analysis • VCS COL 11.5-3, Compliance with 10 CFR Part 50, Appendix I, Sections II.B and II.C for gaseous waste discharges
11.4 Solid Radioactive Waste Management	• None*
11.5 Radiation Monitoring	• VCS COL 11.5-2, QA for effluent and environmental monitoring program • VCS COL 11.5.3, Compliance with 10 CFR Part 50, Appendix I

* This section is entirely IBR or IBR/standard.

Summer FSAR Chapter 12
Radiation Protection

FSAR Section	Site-Specific Evaluations
12.1 Assuring ALARA	• None*
12.2 Radiation Sources	• None*
12.3 Radiation Protection Design Features	• VCS DEP 18.8-1, Relocation of Operations Support Center • VCS SUP 11.2-1, Liquid waste discharge pipe
12.4 Dose Assessment	• VCS SUP12.4-1, Construction worker dose
12.5 Health Physics Facility Design	• VCS DEP 18.8-1, Relocation of Operations Support Center

* This section is entirely IBR or IBR/standard.

Attachment 3

Summer FSAR Chapter 13
Conduct of Operations

FSAR Section	Site-Specific Evaluations
13.1 Organizational Structure of Applicant	• VCS COL 13.1-1 Organization structure • VCS COL 9.5-1 Fire protection • VCS COL 18.6-1 Qualifications of the nuclear plant technical support personnel • VCS COL 18.10-1 Responsibilities of the manager in charge of nuclear training
13.2 Training	• None*
13.3 Emergency Planning	• Presented (1/10/2011)
13.4 Operational Programs	• None*
13.5 Plant Procedures	• VCS SUP 13.5-1 Plant procedures • VCS SUP 13.5-2 Plant procedures
13.7 Fitness for Duty	• None*
13.8 Cyber Security	• None*

* This section is entirely IBR or IBR/standard.

Attachment 3

VC Summer Units 2 and 3 FSAR Chapters 15, 16 and 17

Amy M. Monroe – Licensing
New Nuclear Deployment

Chapter 15
Accident Analysis

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific X/Q values provided in Subsection 2.3.4 are bounded by the values in DCD Section 15A

Chapter 16
Technical Specifications

- DCD Incorporated by Reference
- Standard material incorporated
- Site-specific items are associated with addressing the remaining brackets [] in the AP1000 generic technical specifications.
- Part 4 of the VCSNS COLA reflects the VCSNS Technical Specifications.

Chapter 17
Quality Assurance

- DCD Incorporated by Reference
- Standard material incorporated
- Pre-COL activities are being performed under the existing VCSNS Unit 1 Operational Quality Assurance Plan as supplemented by the New Nuclear Deployment Quality Assurance Plan

Quality Assurance Program Description (QAPD)

- Site-specific QAPD in COLA Part 13 is based on NEI 06-14A Revision 7
- Since all nuclear facilities are located on a single site, the nuclear organization is located primarily on site
- Implementation of the QAPD begins at issuance of the COL for VCSNS Units 2 and 3

Questions?

Attachment 3

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V. C. Summer Units 2 and 3
COL Application Review

Chapters 15, 16, and 17

Accident Analysis,
Technical Specifications, and
Quality Assurance Program

January 10 -11, 2011

Staff Review Team

- Technical Staff

  Chapter 15:

  – Michelle Hart, Siting & Accident Consequences Branch

  Chapter 17:

  – Juan Peralta, Branch Chief, Quality and Vendor Branch

  – Raju Patel, Lead Reviewer, Quality and Vendor Branch

- Project Managers

  – Donald Habib, Chapter 15

  – Sujata Goetz, Chapter 16

  – Tom Galletta, Chapter 17

Summer FSAR Chapter 15

Accident Analysis

FSAR Section	Site-Specific Evaluations
15.0 Accident Analysis	• None*
15.1 Increase in Heat Removal from Primary System	• None*
15.2 Decrease in Heat Removal by the Secondary System	• None*
15.3 Decrease in Reactor Coolant System Flow Rate	• None*
15.4 Reactivity and Power Distribution Anomalies	• None*
15.5 Increase in Reactor Coolant Inventory	• None*
15.6 Decrease in Reactor Coolant Inventory	• None*
15.7 Radioactive Release from a Subsystem or Component	• VCS COL 15.7-1, Consequence of Liquid Waste Tank Failure
15.8 Anticipated Transients without Scram	• None*
15A Evaluation Models and Parameters for Analysis of Radiological Consequences of Accidents	• **VCS COL 2.3-4, DBA Radiological Consequences Analyses**
15B Removal of Airborne Activity from the Containment Atmosphere Following a LOCA	• None*

* This section is entirely IBR or IBR/standard.

Summer FSAR Chapter 16

Technical Specifications

FSAR Section	Site-Specific Evaluations
16.1 Technical Specifications	• VCS COL 16.1-1 related to technical specifications for use as a guide in development of the plant-specific technical specifications.
16.2 Design Reliability Assurance Program	• None*
16.3 Investment Protection	• None*

* This section is entirely IBR or IBR/standard.

Attachment 3

Summer FSAR Chapter 17

Quality Assurance Program

FSAR Section	Site-Specific Evaluations
17.1 Quality Assurance During the Design and Construction Phases	• VCS COL 17.5-1 QAP prior to COL issuance
17.2 Quality Assurance During the Operations Phase	• None*
17.3 Quality Assurance During the Design, Procurement, Fabrication, Inspection, and/or Testing of Nuclear Power Plant Items	• None*
17.4 Design Reliability Assurance Program	• None*
17.5 Quality Assurance Program Description – New License Applicants	• VCS COL 17.5-1 QAP following COL issuance
17.6 Maintenance Rule Program	• None*

*This section is entirely IBR or IBR/Standard

January 10–11, 2011

VCS COL 2.3-4
DBA Radiological Consequences Analyses

- **Issue**
  - Appropriate incorporation by reference of the DBA dose analyses from the AP1000 DCD to thereby show compliance with the offsite dose factors in 10 CFR 52.79(a)(1) and the control room dose criterion in GDC 19.
    - VCS DEP 18.8-1 site-specific TSC (SER 13.3)

- **Resolution**
  - Summer site characteristic short-term atmospheric dispersion (χ/Q) values are bounded by the values given in AP1000 DCD as site parameters. (SER 2.3)
    - Site characteristic χ/Q values are the only site-related DBA dose analysis inputs
    - Dose is directly proportional to the χ/Q values for each time period
    - Summer χ/Qs < AP1000 χ/Qs
    - Summer DBA doses < AP1000 DBA doses
  - AP1000 DCD showed compliance with the offsite and control room dose factors for all DBAs, therefore Summer also complies.

QA Design and Construction Phases

- FSAR Section 17.1
  - Prior to COL issuance - SCE&G is using VCSNS Unit 1, Operational QAP, for oversight of contractors.
  - Staff inspected the program as it is being applied to VCSNS Units 2 and 3 and found it acceptable
    - Staff performed limited scope inspection
    - Identified one violation
    - Applicant has responded to the violation
    - Staff has found the applicant's response acceptable

QA Program Description

- FSAR Section 17.5
  - Following COL issuance – SCE&G will use the QAPD described in the COL FSAR.
    - COL FSAR QAPD is based on NEI Template 06-14, Revision 7.
  - The NRC staff used the requirements of Appendix B to 10 CFR Part 50 and the guidance in SRP Section 17.5 for evaluating the acceptability of the VCSNS COL FSAR Chapter 17.
  - COL Information Item 17.5-1 is addressed in Section 17.5 of the COL FSAR.
  - The staff evaluated the QAPD and concluded:
    - The QAPD complies with the acceptance criteria in SRP Section 17.5 and with the commitments to applicable regulatory guidance.
    - The QAPD provides adequate guidance for the applicant to establish controls that, when properly implemented, complies with Appendix B.

Attachment 3

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Staff Action Items from 1/10/11 Meeting

January 10 – 11, 2011

Action Items From 1/10/10 Meeting

- Provide HABIT Verification Documentation
- Provide a Discussion of How Design Control Document Revision 18 and the Combined License Application Revisions are being reviewed by the staff

Action Items From 1/10/10 Meeting

- HABIT used by staff for confirmatory calculation as an independent check to determine if the staff agrees with the applicant's conclusion that the following chemicals do not pose a threat to control room habitability
  - 28% ammonium hydroxide (Unit 1)
  - Cyclohexylamine (Norfolk Southern rail)
  - Chlorodifluoromethane (Norfolk Southern rail)
- HABIT code endorsed in Regulatory Guide (RG)1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"
  - RG indicates when boiloff or a slow leak is analyzed, the effects of density on vertical diffusion may be considered
  - RG indicates that density effect of heavier-than-air gases should not be considered for releases of a violent nature or for release material that becomes entrained in turbulent air near buildings
- HABIT Code described in NUREG/CR-6210, "Computer Codes for Evaluation of Control Room Habitability (HABIT)"

Attachment 3

Tie between DCD Revision 17 and COL Review