From:	Franzone, Steve <Steve.Franzone@fpl.com>
Sent:	Friday, December 11, 2015 11:22 AM
To:	Comar, Manny
Cc:	Maher, William; Orthen, Richard; TurkeyCOL Resource; Burski, Raymond
Subject:	[External_Sender] Draft License Conditions Comparison between PTN 6 & 7 and VC Summer
Attachments:	20151211_Draft Excel spreadsheet_FPL Lic Cond_revD.xlsx; 20151211_VCS Unit 2 License Amendment 22_DraftCompare.pdf

Manny
I have attached an excel spreadsheet which compares our (PTN 6 & 7) proposed license conditions to VCS Rev 5 COLA proposed license conditions and to the VCS Unit 2 License Conditions (updated to Amendment 22).
We added some extra fields since we were going to be updating our database anyway, however, you can just hide the fields you don't need.

In addition, I have attached a pdf of the VCS Unit 2 License Conditions (updated to Amendment 22). It is marked up to show PTN's proposed LC's against Amendment 22 of the VCS license. Those that are highlighted are word for word. Those highlighted but not exact we used text boxes to try to explain the differences, if minor we noted the difference, if not minor we just said "not word for word". The LCs not highlighted are the ones we do not have a corresponding proposed LC. Again, this is our first cut of what turned out to be a lot more complicated task than I initially thought.

Any questions, please contact us. I will be traveling to WEC/Cranberry next week so feel free to talk with Ray or Rick on the subject if you can't get me right away.
Thanks
Steve Franzone
NNP Licensing Manager - COLA
"You ask what is our aim? I can answer in one word: Victory. Victory at all costs. Victory in spite of all terror. Victory however long and hard the road may be. For without victory there is no survival." ~Winston Churchill
561.904.3793 (office)
754.204.5996 (cell)

Hearing Identifier: TurkeyPoint_COL_Public
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Created By: Steve.Franzone@fpl.com

Recipients:
"Maher, William" <William.Maher@fpl.com>
Tracking Status: None
"Orthen, Richard" <Richard.Orthen@fpl.com>
Tracking Status: None
"TurkeyCOL Resource" <TurkeyCOL.Resource@nrc.gov>
Tracking Status: None
"Burski, Raymond" <RAYMOND.BURSKI@fpl.com>
Tracking Status: None
"Comar, Manny" <Manny.Comar@nrc.gov>
Tracking Status: None

Post Office: GOXSA1809.fplu.fpl.com

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	A	B	C	D	E	F
1	Track Number	Discussion/Subject/Title	DCD Tier 2 Source Location	Implementation Milestone	PTN COLA rev 7 Proposed License Condition	Track Number
2	01	**1. ITAAC (INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA):** There are several ITAAC identified in the COL application. Once incorporated into the COL, the regulations identify the requirements that must be met. The incorporation below includes references to the sensitive unclassified non-safeguards information (including proprietary information) and safeguards information, contained in the AP1000 DCD. Such DCD information is included in this combined license application in the same manner as it is included in the AP1000 DCD, i.e., references in the DCD are included as references in the FSAR, and material incorporated by reference into the DCD is incorporated by reference into the FSAR. Appropriate agreements are in place to provide for the licensee's rights to possession (including constructive possession) and use of the withheld sensitive unclassified non-safeguards information (including proprietary information) and safeguards information referenced in the AP1000 DCD for the life of the project. information) and safeguards information referenced in the AP1000 DCD for the life ofthe project.			The ITAAC identified in the tables in Appendix B are hereby incorporated into this Combined License. After the Commission has made the finding required by 10 CFR 52.103(g), the ITAAC do not constitute regulatory requirements; except for specific ITAAC, which are the subject of a Section 103(a) hearing, their expiration will occur upon final NRC action in such proceeding.	
3	02	**2. COL HOLDER ITEMS:** There are several COL information items that cannot be resolved prior to issuance of the Combined License. The referenced AP1000 design certification has already justified why each COL holder item (as identified in the AP1000 DCD Tier 2 Table 1.8-2) cannot be resolved before the COL is issued, provides sufficient information on these items to support the NRC licensing decision, and identifies an appropriate implementation milestone. Each COL information item that cannot be resolved completely before the COL is issued is also identified as a COL holder item in the FSAR Table 1.8-202. Therefore, in accordance with the guidance in RG 1.206, Section C.III.4.3, the following License Condition is proposed to address these COL holder items. Holder items (per DCD Table 1.8-2) that are addressed by the COL application are not included in the proposed condition. These include COL information item numbers 3.11-1, 9.5-6, 10.1-1, and 13.6-5.			Each COL holder item identified below shall be completed by the identified implementation milestone through completion of the action therein identified.	

	A	B	C	D	E	F
4	02.03.06-01	3.6-1 **As-Designed Pipe Rupture Hazards Analysis**	3.6.4.1	**Prior to installation of the piping and connected components in their final location**	After a Combined License is issued, the following activity will be completed by the COL holder. An as-designed pipe rupture hazard evaluation will be available for NRC review. The completed as-designed pipe rupture hazards evaluation will be in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. Systems, structures, and components identified to be essential targets and appropriate mitigation features (Reference is DCD Table 3.6-3) will be confirmed as part of the evaluation, and updated information will be provided as appropriate. A pipe rupture hazards analysis is part of the piping design. The evaluation will be performed for high and moderate energy piping to confirm the protection of systems, structures, and components (SSCs), which are required to be functional during and following a design basis event. The locations of the postulated ruptures and essential targets will be established and required pipe whip restraints and jet shield designs will be included. The evaluation will address environmental and flooding effects of cracks in high and moderate energy piping. The as-designed pipe rupture hazards evaluation is prepared on a generic basis to address COL applications referencing the AP1000 design.	
5	02.03.07-03	3.7-3 **Seismic Interaction Review**	3.7.5.3	**Prior to initial fuel load**	The seismic interaction review will be updated by the Combined License holder for as-built information. This review is performed in parallel with the seismic margin evaluation. The review is based on as-procured data, as well as the as-constructed condition. The as-built seismic interaction review is not provided with the COL application, but is completed prior to fuel load.	
6	02.03.07-04	3.7-4 **Reconcillation of Seismic Analyses of Nuclear Island Structures**	3.7.5.4	**Prior to initial fuel load**	The Combined License holder will reconcile the seismic analyses described in Subsection 3.7.2 for detail design changes, such as those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information. Deviations are acceptable based on an evaluation consistent with the methods and procedure of Section 3.7 provided the amplitude of the seismic floor response spectra, including the effect due to these deviations, does not exceed the design basis floor response spectra by more than 10 percent. The Combined License holder will complete this reconciliation prior to fuel load.	
7	02.03.09-07	3.9-7 **As-Designed Piping Analysis**	3.9.8.7	**Prior to installation of the piping and connected components in their final location**	After a Combined License is issued, the following activity will be completed by the COL holder: The as-designed piping analysis is provided for the piping lines chosen to demonstrate all aspects of the piping design. A design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class 1 piping using the methods and criteria outlined in DCD Table 3.9-19 is made available for NRC review. The availability of the piping design information and design reports for the piping packages is identified to the NRC.	

	A	B	C	D	E	F
8	02.04.04-2	4.4-2 **Confirm Assumptions for Safety Analyses DNBR Limits**	4.4.7	**Prior to initial fuel load**	Combined License applicants referencing the AP1000 certified design will address changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, or initial core design from that presented in the DCD. Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in Subsection 7.1.6, Combined License applicants will calculate the design limit DNBR values using the RTDP with these instrumentation uncertainties and confirm that either the design limit DNBR values as described in Section 4.4 remain valid, or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties, such as rod bow penalty.	
9	02.05.03-1	5.3-1 **Reactor Vessel Pressure —** **Temperature Limit Curves**	5.3.6.1	**Prior to initial fuel load**	The COL Holder shall update the P/T limits using the PTLR methodologies approved in the AP1000 DCD using the plant-specific material properties or confirm that the reactor vessel material properties meet the specifications and use the Westinghouse generic PTLR curves.	
10	02..05.03-4	5.3-4 **Reactor Vessel Materials** **Properties Verification**	5.3.6.4.1	**Prior to initial fuel load**	The Combined License holder will complete prior to fuel load verification of plant-specific belt line material properties consistent with the requirements in Subsection 5.3.3.1 and Tables 5.3-1 and 5.3-3. The verification will include a pressurized thermal shock evaluation based on as-procured reactor vessel material data and the projected neutron fluence for the plant design objective of 60 years. This evaluation report will be submitted for NRC staff review.	
11	02.09.01-7	9.1-7 **Coupon Monitoring Program**	9.1.6	**Prior to commercial operation**	A spent fuel rack Metamic coupon monitoring program will be implemented when the plant is placed into commercial operation. This program will include tests to monitor bubbling, blistering, cracking, or flaking; and a test to monitor for corrosion, such as weight loss measurements and/or visual examination. The program will also include testing to monitor changes in physical properties of the absorber material, including neutron attenuation and thickness measurements.	
12	02.10.02-1	10.2-1 **Turbine Maintenance and** **Inspection**	10.2.6	**Prior to initial fuel load**	The Combined License holder will submit to the NRC staff for review prior to fuel load, and then implement a turbine maintenance and inspection program. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in Subsection 10.2.3.6. The Combined License holder will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis after the fabrication of the turbine and prior to fuel load.	

	A	B	C	D	E	F
13	02.13.06-5	13.6-5 **Cyber Security Program**	13.6.1	**Prior to initial fuel load**	The Combined License holder will develop and implement a cyber security program prior to initial fuel load.	
14	02.14.04-2	14.4-2 **Test Specifics and Procedures**	14.4.2	**Prior to initial fuel load**	NOTE — addressed by proposed License Condition #6.	
15	02.14.04-3	14.4-3 **Conduct of Test Program**	14.4.3		NOTE — addressed by proposed License Conditions #3 and #6.	
16	02.14.04-4	14.4-4 **Review and Evaluation of Test Results**	14.4-4		NOTE — addressed by proposed License Condition #9.	
17	02.14.4-6	14.4-6 **First-Plant-Only and Three-Plant-Only Tests**	14.4.6		NOTE — addressed by proposed License Conditions #7 and #9.	
18	02.15.0-1	15.0-1 **Documentation of Plant Calorimetric Uncertainty Methodology**	15.0.15.1		NOTE — addressed by proposed ITAAC Table 2.5.4-2, item 4.	

	A	B	C	D	E	F
19	02.19.59.10-1	19.59.10-1 **As-Built SSC HCLPF Comparison to Seismic Margin Evaluation**	19.59.10.5	**Prior to initial fuel load**	The Combined License holder referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 seismic margins analysis prior to fuel load. A verification walkdown will be performed with the purpose of identifying differences between the as-built plant and the design. Any differences will be evaluated and the seismic margins analysis modified as necessary to account for the plant specific-design, and any design changes or departures from the certified design. Spacial interactions are addressed by COL information item 3.7-3. Details of the process will be developed by the Combined License holder. The Combined License holder referencing the AP1000 certified design should compare the as-built SSC HCLPFs to those assumed in the AP1000 seismic margin evaluation prior to fuel load. Deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis should be evaluated to determine if vulnerabilities have been introduced. The requirements to which the equipment is to be purchased are included in the equipment specifications. Specifically, the equipment specifications include: 1. Specific minimum seismic requirements consistent with those used to define the Table 19.55-1 HCLPF values. This includes the known frequency range used to define the HCLPF by comparing the required response spectrum (RRS) and test response spectrum (TRS). The test response spectra are chosen so as to demonstrate that no more than one percent rate of failure is expected when the equipment is subjected to the applicable seismic margin ground motion for the equipment identified to be applicable in the seismic margin insights of the site-specific PRA. The range of frequency response that is required for the equipment with its structural support is defined. 2. Hardware enhancements that were determined in previous test programs and/or analysis programs will be implemented.	
20	02.19.59.10-2	19.59.10-2 **Evaluation of As-Built Plant Versus Design in AP1000 PRA and Site-Specific PRA External Events**	19.59.10.5	**Prior to initial fuel load**	The Combined License holder referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 PRA and Table 19.59-18 prior to fuel load. The plant specific PRA-based insight differences will be evaluated and the plant specific PRA model modified as necessary to account for the plant specific-design and, any design changes or departures from the design certification PRA.	
21	02.19.59.10-3	19.59.10-3 **Internal Fire and Internal Flood Analyses**	19.59.10.5	**Prior to initial fuel load**	The Combined License holder referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analyses prior to fuel load. Plant specific internal fire and internal flood analyses will be evaluated and the analyses modified as necessary to account for the plant-specific design, and any design changes or departures from the certified design.	
22	02.19.59.10-4	19.59.10-4 **Implement Severe Accident Management Guidance**	19.59.10.5	**Prior to startup testing**	NOTE — addressed by proposed License Condition #6.	

	A	B	C	D	E	F
23	02.19.59.10-5	19.59.10-5 **Equipment Survivability**	19.59.10.5	**Prior to initial fuel load**	The Combined License holder referencing the AP1000 certified design will perform a thermal lag assessment of the as-built equipment listed in Tables 6b and 6c in Attachment A of APP-GW-GLR-069 to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. This assessment is performed prior to fuel load and is required only for equipment used for severe accident mitigation that has not been tested at severe accident conditions. The Combined License holder will assess the ability of the as-built equipment to perform during severe accident hydrogen burns using the Environment Enveloping method or the Test Based Thermal Analysis method discussed in EPRI NP-4354.	
24	03	**3. OPERATIONAL PROGRAM IMPLEMENTATION:** The provisions of the regulations address implementation milestones for some operational programs. The NRC will use license conditions to ensure implementation for those operational programs whose implementation is not addressed in the regulations. FSAR Subsection 13.4, Table 13.4-201, identifies several programs required by regulations that must be implemented by a milestone to be identified in a license condition.			The licensee shall implement the programs or portions of programs identified below on or before the associated milestones identified below.	03
25	03.A				A. Construction Initiation — The licensee shall implement each operational program identified below prior to initiating construction of nuclear safety- or security-related structures, systems, or components: None identified.	03.A
26	03.B			18 Months Before Fuel Load	B. 18 Months Before Fuel Load — The licensee shall implement each operational program identified below at least 18 months prior to scheduled date of initial fuel load: B.1 – Reactor Operator Training	03.B
27	03.B.1	B.1 – Reactor Operator Training		18 Months Before Fuel Load	B.1 – Reactor Operator Training	03.B.1

	A	B	C	D	E	F
28	03.C			Receipt Of Materials	C. Receipt Of Materials — The licensee shall implement each operational program identified below prior to initial receipt of by-product, source, or special nuclear materials on site (excluding Exempt Quantities as described in 10 CFR 30.18): C.1 – Radiation Protection (applicable portions) C.2 – Fire Protection Program (applicable portions) C.3 – Non Licensed Plant Staff Training Program (applicable portions) C.4 – Deleted C.5 – Deleted C.6 – SNM Material Control and Accounting Program	03.C
29	03.C.1	C.1 — Radiation Protection (applicable portions)		Receipt Of Materials	C.1 – Radiation Protection (applicable portions)	
30	03.C.2	C.2 – Fire Protection Program (applicable portions)		Receipt Of Materials	C.2 – Fire Protection Program (applicable portions)	03.C.2
31	03.C.3	C.3 – Non Licensed Plant Staff Training Program (applicable portions)		Receipt Of Materials	C.3 – Non Licensed Plant Staff Training Program (applicable portions)	03.C.3
32	03.C.4				C.4 – Deleted	03.C.4
33	03.C.5				C.5 – Deleted	03.C.5

	A	B	C	D	E	F
34	03.C.6	C.6 – SNM Material Control and Accounting Program		Receipt Of Materials	C.6 – SNM Material Control and Accounting Program	03.C.6
35	03.D			Fuel Receipt	D. Fuel Receipt — The licensee shall implement each operational program identified below prior to initial receipt of fuel onsite:	03.D.1
36	03.D.1	D.1 – Fire Protection (applicable portions)		Fuel Receipt	D.1 – Fire Protection (applicable portions)	03.D.1
37	03.D.2	D.2 – Radiation Protection (applicable portions)			D.2 – Radiation Protection (applicable portions)	03.D.2
38	03.D.3	D.3 – Special Nuclear Material Physical Protection Program		Fuel Receipt	D.3 – Special Nuclear Material Physical Protection Program	03.D.3
39	03.D.4			Fuel Receipt	D.4 – Deleted	03.D.4
40	03.E			prior to initial construction testing:	E. Construction Testing — The licensee shall implement each operational program identified below prior to initial construction testing.	03.E
41	03.E.1			prior to initial construction testing:	E.1 – Initial Test Program — Construction Testing	03.E.1
42	03.E.2	NA				03.E.2
43	03.F			prior to initial preoperational testing	E. Preoperational Testing — The licensee shall implement each operational program identified below prior to initial preoperational testing: F.1 – Initial Test Program — Preoperational Testing	03.F

	A	B	C	D	E	F
44	03.F.1	F.1 – Initial Test Program — Preoperational Testing		prior to initial preoperational testing	F.1 – Initial Test Program — Preoperational Testing	03.F.1
45	03.G			prior to initial fuel load	G. Fuel Loading — The licensee shall implement each operational program identified below prior to initial fuel load:	03.G
46	03.G.01	G.1 – Environmental Qualification		prior to initial fuel load	G.1 – Environmental Qualification	03.G.01
47	03.G.02	G.2 – Pre-Service Testing		prior to initial fuel load	G.2 – Pre-Service Testing	03.G.02
48	03.G.03	G.3 – Process and Effluent Monitoring and Sampling		prior to initial fuel load	G.3 – Process and Effluent Monitoring and Sampling	03.G.03

	A	B	C	D	E	F
49	03.G.04	G.4 – Radiation Protection (applicable portions)		prior to initial fuel load	G.4 – Radiation Protection (applicable portions)	03.G.04
50	03.G.05	G.5 – Motor-Operated Valve Testing		prior to initial fuel load	G.5 – Motor-Operated Valve Testing	03.G.05
51	03.G.06	G.6 – Fire Protection		prior to initial fuel load	G.6 – Fire Protection	03.G.06
52	03.G.07				G.7 – Deleted	03.G.07
53	03.G.08	G.8 – Containment Leakage Rate Testing Program		prior to initial fuel load	G.8 – Containment Leakage Rate Testing Program	03.G.08
54	03.G.09	G.9 – Physical Security		prior to initial fuel load	G.9 – Physical Security	03.G.09
55	03.G.10	G.10 – Cyber Security		prior to initial fuel load	G.10 – Cyber Security	03.G.10
56	03.H			prior to initial startup testing	H. Startup Testing — The licensee shall implement each operational program identified below prior to initial startup testing:	03.H

	A	B	C	D	E	F
57	03.H.01	H.1 – Initial Test Program — Startup Testing		prior to initial startup testing	H.1 – Initial Test Program — Startup Testing	03.H.1
58	03.I				I. MODE 4 – Not used	03.I
59	03.J			prior to initial criticality	J. Initial Criticality — The licensee shall implement each operational program identified below prior to initial criticality:	03.J
60	03.J.01	J.1 – Reactor Vessel Material Surveillance		prior to initial criticality	J.1 – Reactor Vessel Material Surveillance	03.J.1
61	03.K			prior to initial radioactive waste shipment	Waste Shipment — The licensee shall implement each operational program identified below prior to initial radioactive waste shipment:	03.K
62	03.K.1			prior to initial radioactive waste shipment	K.1 – Radiation Protection	03.K.1
63	04	**NOT USED**				
64	05	**5. SECURITY PROGRAM:**				
65	05A	**A. SECURITY PROGRAM IMPLEMENTATION** An implementation license condition approved in the staff requirements memo regarding SECY-05-0197 applies to the security program.		nuclear fuel is onsite (protected area), and continuing until all nuclear fuel is permanently removed from the site.	The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan and cyber security plan, and all amendments made pursuant to the authority of 10 CFR 50.90, 50.54(p), 52.97, and Section VIII of Appendix D to Part 52 when nuclear fuel is onsite (protected area), and continuing until all nuclear fuel is permanently removed from the site.	05A
66	05B	**B. SPECIAL NUCLEAR MATERIAL PHYSICAL PROTECTION** A license condition is proposed to address when the boundary for physical protection of new fuel as SNM is required to be extended from the controlled access area (CAA) in accordance with the requirements of 10 CFR 73.67 to the operational protected area (PA) in accordance with 10 CFR 73.55.		new fuel as SNM in a controlled access area (CAA) in accordance with the requirements of 10 CFR 73.67, until such time as an operational protected area (PA)	The licensee shall receive and store new fuel as SNM in a controlled access area (CAA) in accordance with the requirements of 10 CFR 73.67, until such time as an operational protected area (PA) that satisfies the requirements of 10 CFR 73.55(e)(8) is established. If new fuel is already stored in a CAA that is within the boundary of the proposed PA, then upon declaration of an operational PA, the remaining requirements of 10 CFR 73.55 shall be implemented. The PA shall be established and declared operational prior to initial fuel load.	05B

	A	B	C	D	E	F
67	06	**6. OPERATIONAL PROGRAM READINESS:** The NRC inspection of operational programs will be the subject of the following license condition in accordance with SECY-05-0197:			The licensee shall submit to the appropriate director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspections of operational programs listed in the operational program FSAR Table 13.4-201. The schedule shall be updated every six months until 12 months before scheduled fuel loading, and every month thereafter until either the operational programs in the FSAR table have been fully implemented or the plant has been placed in commercial service, whichever comes first. This schedule shall also address:	
68	06				a. the emergency planning implementation procedures to the NRC consistent with 10 CFR Part 50, Appendix E, Section V. b. the implementation of site specific Severe Accident Management Guidance. c. a reactor vessel pressurized thermal shock evaluation at least 18 months prior to initial fuel load. d. the approved preoperational and startup test procedures (including the sitespecific startup administration manual (procedure) prior to initiating the plant initial test program) in accordance with FSAR Subsection 14.2.3. e. an emergency response data system (ERDS) implementation program plan consistent with 10 CFR Part 50, Appendix E, Section V.	
69					f. a flow accelerated corrosion (FAC) program implementation schedule, including the construction phase activities. g. full implementation of the operational and programmatic elements of responding to an event associated with a loss of large areas of the plant due to explosions or fire, prior to initial fuel load. h. the spent fuel rack Metamic coupon monitoring program implementation. i. the implementation of construction and inspection procedures for steel concrete composite (SC) construction activities for seismic Category I nuclear island modules (including shield building SC modules) before and after concrete placement, and inspection of such construction before and after concrete placement. j. the availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty, prior to initial fuel load. k. the availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation, prior to initial fuel load.	

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71	07	**7. FIRST-PLANT-ONLY AND FIRST-THREE-PLANT-ONLY TESTING:** Certain design features of the AP1000 plant will be subjected to special tests to establish unique phenomenological performance parameters of the AP1000 design. Because of the standardization of the AP1000 design, these special tests (designated as first-plant-only tests and first-three-plant-only tests) are not required on subsequent plants. Once these tests are completed by the first plant (or first three plants) and appropriate documentation identified, the subsequent plants need only reference the applicable documentation to show that the first plant (or first three plants) completed the required testing. Accordingly, the following license condition is proposed:			First-Plant-Only and First-Three-Plant-Only Testing A licensee shall provide written identification of the applicable references for documentation for the completion of the testing to the Director of the Office of New Reactors (or equivalent NRC management) within thirty (30) calendar days of the licensee confirmation of acceptable test results. Subsequent plant licensees crediting completion of testing by the first-plant or by the first-threeplants shall provide a report referencing the applicable documentation identified by the first (or first three) plant(s) confirming the testing to the Director of the Office of New Reactors (or equivalent NRC management). This report shall be provided to NRC either prior to initiation of pre-operational testing, or within sixty (60) days of the identification of the documentation for the completion of the testing by the first plant (or third plant, as appropriate), whichever is later.	
72	08	**8. STARTUP TESTING:** FSAR Section 14.2 specifies certain startup tests that must be completed after fuel load. Operating licenses typically have included the following condition related to startup testing			Any changes to the Initial Startup Test Program described in Chapter 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 or Section VIII of Appendix D to 10 CFR Part 52 shall be reported in accordance with 10 CFR 50.59(d) within one month of such change	
73	09	**STARTUP PROGRAM TEST RESULTS:** Certain milestones within the startup testing phase of the initial test program (i.e., pre-critical testing, criticality testing, and low-power (<5% RTP) testing) are controlled through license conditions to ensure that relevant test results are reviewed, evaluated, and approved by the designated licensee management before proceeding with the power ascension test phase.			Accordingly, the following license conditions are proposed:	

	A	B	C	D	E	F
74	09.01				Pre-operational Testing Following completion of pre-operational testing, the licensee shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed.	
75	09.02				Pre-critical and Criticality Testing 1. Following completion of pre-critical and criticality testing, the licensee shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed. 2. The licensee shall provide written notification to the Director of the Office of New Reactors (or equivalent NRC management) within fourteen (14) calendar days of completion of the pre-critical and criticality testing. 2. The licensee shall provide written notification to the Director of the Office of New Reactors (or equivalent NRC management) within fourteen (14) calendar days of completion of the low power testing.	

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76	09.03				Low-Power (<5% RTP) Testing 1. Following completion of low-power (<5% RTP) testing, the licensee shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed.	
77	09.04				At-Power (5%–100% RTP) Testing 1. Following completion of at-power testing (at or above 5% RTP up to and including testing at 100% RTP), the licensee shall review and evaluate individual test results. Test exceptions or results which do not meet acceptance criteria are identified to the affected and responsible organizations, and corrective actions and retests, as required, are performed. 2. The licensee shall provide written notification to the Director of the Office of New Reactors (or equivalent NRC management) within fourteen (14) calendar days of completion of the at-power testing.	

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78	10	**ENVIRONMENTAL PROTECTION PLAN:** Operating licenses typically have included the following condition related to environmental protection.			The issuance of this COL, subject to the Environmental Protection Plan (EPP) and the conditions for the protection of the environment set forth herein, is in accordance with the National Environmental Policy Act of 1969, as amended, and with applicable sections of 10 CFR Part 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions, as referenced by Subpart C of 10 CFR Part 52, Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants, and all applicable requirements therein have been satisfied.	
79	11	**EMERGENCY PLANNING ACTIONS:**				11
80	11A			at least 180 days prior to initial fuel load	A. The licensee shall submit a fully developed set of site-specific Emergency Action levels (EALs) to the NRC in accordance with the NRC-endorsed version of NEI 07-01, Revision 0, with no deviations. The EALs shall have been discussed and agreed upon with stat and local officials. These fully developed EALs shall be submitted to the NRC for confirmation at least 180 days prior to initial fuel load.	11
81	11B			At least two (2) years before scheduled initial fuel load	B. At least two (2) years before scheduled initial fuel load, the licensee shall have performed an assessment of emergency response staffing in accordance with NEI 10-05, Assessment of On-Shift Emergency Response Organization Staffing and Capabilities, or other NRC-endorsed guidance in effect six (6) months prior to commencement of the assessment.	
82						
83	12	**FUKUSHIMA ACTIONS:**				
84	12.A				**A. MITIGATION STRATEGIES** {Prior to initial fuel load, the licensee shall fully implement the following actions associated with mitigation strategies including procedures, guidance, training, and acquisition, staging, or installing of equipment needed for the strategies: 1. Develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment and spent fuel pool (SFP) cooling capabilities following a beyonddesign basis external event. These strategies must: Be capable of mitigating a simultaneous loss of all ac power and loss of normal access to the normal heat sink, and Have adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on the Turkey Point Units 6 & 7 site, and Have the capability to be implemented in all modes. 2. Provide reasonable protection for the associated equipment from external events. Such protection must demonstrate that there is adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on the Turkey Point Units 6 & 7 site. 3. The licensee shall within one (1) year after issuance of the COL, submit to the NRC for review an overall integrated plan, including a description of how compliance with the requirements described in this license condition will be achieved. 4. The licensee shall provide to the NRC an initial status report sixty (60) days following issuance of the COL and at six (6) month intervals following submittal of the overall integrated plan described above which delineates progress made in implementing the requirements of this license condition.}	
85	12.B				**B. RELIABLE SPENT FUEL POOL LEVEL INSTRUMENTATION** {Prior to initial fuel load, the licensee shall fully implement the following requirements for spent fuel pool (SFP) level indication: The spent fuel pool instrumentation shall be maintained available and reliable through the development and implementation of a training program. The training program shall include provisions to ensure trained personnel can route the temporary power lines from the alternate power source to the appropriate connection points and connect the alternate power source to the	

	A	B	C	D	E	F
86	12.C				**C. EMERGENCY PLANNING ACTIONS** Staffing At least two (2) years prior to scheduled initial fuel load, the licensee shall have performed an assessment of the onsite and augmented staffing capability to satisfy the regulatory requirements for response to a multi-unit event. The staffing assessment will be performed in accordance with NEI 12-01, Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities, or other NRC-endorsed guidance in effect six (6) months prior to commencement of the assessment. At least two (2) years prior to scheduled initial fuel load, the licensee will revise the Emergency Plan to include the following: Incorporation of corrective actions identified in the staffing assessment described above. Identification of how the augmented staff will be notified given degraded communications capabilities. Communications At least two (2) years prior to scheduled fuel load, the licensee shall have performed an assessment of on-site and off-site communications systems and equipment required during an emergency event to ensure communications capabilities can be maintained during prolonged station blackout conditions. The communications capability assessment will be performed in accordance with NEI 12-01, Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities, or other NRC-endorsed guidance in effect six (6) months prior to commencement of the assessment. At least one hundred eighty (180) days prior to scheduled initial fuel load, the licensee shall complete implementation of corrective actions identified in the communications capability assessment described above, including any related emergency plan and implementing procedure changes and associated training.	
87	13	**RADWASTE BUILDING RADIOACTIVITY LIMITS:**			Prior to initial fuel load, the licensee shall develop, implement, and maintain procedural controls limiting radionuclide inventory in each of the Radwaste Building Monitor Tanks, and separately in each of up to three (3) Radwaste Building mobile radwaste processing systems to below A2 quantities for radionuclides specified in Appendix A to 10 CFR Part 71 (Tables A-1 and A-3), as described in FSAR Section 13.5.2.2.5. The procedures shall also ensure that any additional equipment located in the Radwaste Building is limited to the A2 quantities and that the total cumulative radioactive inventory contained in unpackaged wastes (including liquid waste, wet waste, solid waste, gaseous waste, activated or contaminated metals and components, and contaminated waste present at any time in the Radwaste Building) is limited so that an unmitigated release, occurring over a two hour time period, would not result in a dose of greater than 100 millirem at the protected area boundary or an unmitigated exposure, occurring over a two hour time period, would not result in a dose of greater than 5 rem to site personnel located 10 feet from the total cumulative radioactive inventory	

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1	Discussion/Subject/Title	DCD Tier 2 Source Location	Implementation Milestone	VCS COLA rev 5 Proposed License Condition	Track Number	VCS Unit 2 COL Amendment 22 License Condition
2				The ITAAC identified in the tables in Appendix B are hereby incorporated into this Combined License. After the Commission has made the finding required by 10CFR 52.103(g), the ITAAC do not constitute regulatory requirements; except for specific ITAAC, which are the subject of a Section 103(a) hearing, their expiration will occur upon final Commission action in such proceeding.		(8) The Technical Specifications, Environmental Protection Plan, and ITAAC in Appendices A, B, and C, respectively of this license, as revised through Amendment No. 22, are hereby incorporated into this license.
3	2. COL HOLDER ITEMS: There are several COL information items that cannot be resolved prior to issuance of the Combined License. The referenced AP1000 design certification has already justified why each COL holder item (as identified in the AP1000 DCD Tier 2 Table 1.8-2) cannot be resolved before the COL is issued, provides sufficient information on these items to support the NRC licensing decision, and identifies an appropriate implementation milestone. Each COL information item that cannot be resolved completely before the COL is issued is also identified as a COL holder item in the FSAR Table 1.8-202. Therefore, in accordance with the guidance in RG 1.206, Section C.III.4.3, the following License Condition is proposed to address these COL holder items. Holder items (per DCD Table 1.8-2) that are addressed by the COL application are not included in the proposed condition. These include COL information item numbers 3.11-1, 9.5-6, 10.1-1, and 13.6-5.			Each COL holder item identified below shall be completed by the identified implementation milestone through completion of the action therein identified.		

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4	Each COL holder item identified below shall be completed by the identified implementation milestone through completion of the action therein identified.			After a Combined License is issued, the following activity will be completed by the COL holder. An as-designed pipe rupture hazard evaluation will be available for NRC review. The completed as-designed pipe rupture hazards evaluation will be in accordance with the criteria outlined in DCD Subsections 3.6.1.3.2 and 3.6.2.5. Systems, structures, and components identified to be essential targets and appropriate mitigation features (Reference is DCD Table 3.6-3) will be confirmed as part of the evaluation, and updated information will be provided as appropriate. A pipe rupture hazards analysis is part of the piping design. The evaluation will be performed for high and moderate energy piping to confirm the protection of systems, structures, and components (SSCs), which are required to be functional during and following a design basis event. The locations of the postulated ruptures and essential targets will be established and required pipe whip restraints and jet shield designs will be included. The evaluation will address environmental and flooding effects of cracks in high and moderate energy piping. The as-designed pipe rupture hazards evaluation is prepared on a generic basis to address COL applications referencing the AP1000 design.		(12) Site- and Unit-specific Conditions (a) Before commencing installation of individual piping segments and connected components in their final locations, SCE&G shall complete the as-designed pipe rupture hazards analysis for compartments (rooms) containing those segments in accordance with the criteria outlined in the AP1000 DCD, Rev. 19, Sections 3.6.1.3.2 and 3.6.2.5, and shall inform the Director of NRO, or the Director's designee, in writing, upon the completion of this analysis and the availability of the as-designed pipe rupture hazards analysis reports.
5	**3.7-3** **Seismic Interaction Review**			The seismic interaction review will be updated by the Combined License holder for as-built information. This review is performed in parallel with the seismic margin evaluation. The review is based on as-procured data, as well as the as-constructed condition. The as-built seismic interaction review is not provided with the COL application, but is completed prior to fuel load.		(12) Site- and Unit-specific Conditions (f) Before initial fuel load, SCE&G shall: 1. Update the seismic interaction analysis in AP1000 DCD, Rev. 19, Section 3.7.3.5 to reflect as-built information, which must be based on as-procured data, as well as the as-constructed condition;
6	**3.7-4** **Reconcillation of Seismic Analyses of Nuclear Island Structures**			The Combined License holder will reconcile the seismic analyses described in Subsection 3.7.2 for detail design changes, such as those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information. Deviations are acceptable based on an evaluation consistent with the methods and procedure of Section 3.7 provided the amplitude of the seismic floor response spectra, including the effect due to these deviations, does not exceed the design basis floor response spectra by more than 10 percent. The Combined License holder		(12) Site- and Unit-specific Conditions (f) Before initial fuel load, SCE&G shall: 2. Reconcile the seismic analyses described in Section 3.7.2 of the AP1000 DCD, Rev. 19, to account for detailed design changes, including, but not limited to, those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information;
7	**3.9-7** **As-Designed Piping Analysis**			After a Combined License is issued, the following activity will be completed by the COL holder: The as-designed piping analysis is provided for the piping lines chosen to demonstrate all aspects of the piping design. A design report referencing the as-designed piping calculation packages, including ASME Section III piping analysis, support evaluations and piping component fatigue analysis for Class 1 piping using the methods and criteria outlined in DCD Table 3.9-19 is made available for NRC review. The availability of the piping design information and design		(12) Site- and Unit-specific Conditions (b) Before commencing installation of individual piping segmentsidentified in AP1000 DCD, Rev. 19, Section 3.9.8.7, and connectedcomponents in their final locations in the facility, SCE&G shall complete the analysis of the as-designed individual piping segmentsand shall inform the Director of NRO, or the Director's designee, inwriting, upon the completion of these analyses and the availability of the design reports for the selected piping packages.

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8	**4.4-2** **Confirm Assumptions for Safety Analyses DNBR Limits**			Combined License applicants referencing the AP1000 certified design will address changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, or initial core design from that presented in the DCD. Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in Subsection 7.1.6, Combined License applicants will calculate the design limit DNBR values using the RTDP with these instrumentation uncertainties and confirm that either the design limit DNBR values as described in Section 4.4 remain valid, or that the safety analysis minimum DNBR bounds the		(12) Site- and Unit-specific Conditions (f) Before initial fuel load, SCE&G shall: 3. Calculate the instrumentation uncertainties of the actual plant operating instrumentation to confirm that either the design limit departure from nucleate boiling ratio (DNBR) values remain valid or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties;
9	**5.3-1** **Reactor Vessel Pressure —** **Temperature Limit Curves**			The COL Holder shall update the P/T limits using the PTLR methodologies approved in the AP1000 DCD using the plant-specific material properties or confirm that the reactor vessel material properties meet the specifications and use the Westinghouse generic PTLR curves.		(12) Site- and Unit-specific Conditions (f) Before initial fuel load, SCE&G shall: 4. Update the pressure-temperature (P-T) limits using the pressure temperature limits report (PTLR) methodologies approved in AP1000 DCD, Rev. 19, using the plant-specific material properties or confirm that the reactor vessel material properties meet the specifications of and use the Westinghouse generic PTLR curves;
10	**5.3-4** **Reactor Vessel Materials** **Properties Verification**			The Combined License holder will complete prior to fuel load verification of plant-specific belt line material properties consistent with the requirements in Subsection 5.3.3.1 and Tables 5.3-1 and 5.3-3. The verification will include a pressurized thermal shock evaluation based on as-procured reactor vessel material data and the projected neutron fluence for the plant design objective of 60 years. This evaluation report will be submitted for NRC staff review.		(12) Site- and Unit-specific Conditions (f) Before initial fuel load, SCE&G shall: 5. Verify that plant-specific belt line material properties are consistent with the properties given in AP1000 DCD Rev. 19, Section 5.3.3.1 and Tables 5.3-1 and 5.3-3. The verification must include a pressurized thermal shock (PTS) evaluation based on as-procured reactor vessel material data and the projected neutron fluence for the plant design objective. Submit this PTS evaluation report to the Director of NRO, or the Director's designee, in
11	**9.1-7** **Coupon Monitoring Program**			A spent fuel rack Metamic coupon monitoring program will be implemented when the plant is placed into commercial operation. This program will include tests to monitor bubbling, blistering, cracking, or flaking; and a test to monitor for corrosion, such as weight loss measurements and/or visual examination. The program will also include testing to monitor changes in physical properties of the absorber material,		(12)(e) Site- and Unit-specific Conditions 2. The spent fuel rack Metamic Coupon monitoring program (before initial fuel load);
12	**10.2-1** **Turbine Maintenance and** **Inspection**			The Combined License holder will submit to the NRC staff for review prior to fuel load, and then implement a turbine maintenance and inspection program. The program will be consistent with the maintenance and inspection program plan activities and inspection intervals identified in Subsection 10.2.3.6. The Combined License holder will have available plant-specific turbine rotor test data and calculated toughness curves that support the material property assumptions in the turbine rotor analysis after the fabrication of the turbine and prior to fuel load.		(12)(e) Site- and Unit-specific Conditions 4. A turbine maintenance and inspection program, which must be consistent with the maintenance and inspection program plan activities and inspection intervals identified in FSAR Section 10.2.3.6 (before initial fuel load);

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13	**13.6-5** **Cyber Security Program**			The Combined License holder will develop and implement a cyber security program prior to initial fuel load.		
14	**14.4-2** **Test Specifics and Procedures**			NOTE — addressed by proposed License Condition #6.		
15	**14.4-3** **Conduct of Test Program**			NOTE — addressed by proposed License Conditions #3 and #6.		
16	**14.4-4** **Review and Evaluation of Test Results**			NOTE — addressed by proposed License Condition #9.		
17	**14.4-6** **First-Plant-Only and Three-Plant-Only Tests**			NOTE — addressed by proposed License Conditions #7 and #9.		
18	**15.0-1** **Documentation of Plant Calorimetric Uncertainty Methodology**			NOTE — addressed by proposed ITAAC Table 2.5.4-2, item 4.		(12)(e) Site- and Unit-specific Conditions 5. The availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty (before initial fuel load); 6. The availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation (before initial fuel load); changes or departures from the certified design. SCE&G shall compare the as-built structures, systems, and components (SSC) high confidence, low probability of

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19	**19.59.10-1** **As-Built SSC HCLPF** **Comparison to Seismic Margin** **Evaluation**			The Combined License holder referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 seismic margins analysis prior to fuel load. A verification walkdown will be performed with the purpose of identifying differences between the as-built plant and the design. Any differences will be evaluated and the seismic margins analysis modified as necessary to account for the plant specific-design, and any design changes or departures from the certified design. Spacial interactions are addressed by COL information item 3.7-3. Details of the process will be developed by the Combined License holder. The Combined License holder referencing the AP1000 certified design should compare the as-built SSC HCLPFs to those assumed in the AP1000 seismic margin evaluation prior to fuel load. Deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis should be evaluated to determine if vulnerabilities have been introduced. The requirements to which the equipment is to be purchased are included in the equipment specifications. Specifically, the equipment specifications include: 1. Specific minimum seismic requirements consistent with those used to define the Table 19.55-1 HCLPF values. This includes the known frequency range used to define the HCLPF by comparing the required response spectrum (RRS) and test response spectrum (TRS). The test response spectra are chosen so as to demonstrate that no more than one percent rate of		(12) Site- and Unit-specific Conditions (f) Before initial fuel load, SCE&G shall: 6. Review differences between the as-built plant and the design used as the basis for the AP1000 seismic margin analysis. SCE&G shall perform a verification walkdown to identify differences between the as-built plant and the design. SCE&G shall evaluate any differences and must modify the seismic margin analysis as necessary to account for the plant-specific design and any design changes or departures from the certified design. SCE&G shall compare the as-built structures, systems, and components (SSC) high confidence, low probability of failures (HCLPFs) with those assumed in the AP1000 seismic margin evaluation, before initial fuel load. SCE&G shall evaluate deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis to determine if vulnerabilities have been introduced;
20	**19.59.10-2** **Evaluation of As-Built Plant** **Versus Design in AP1000 PRA** **and Site-Specific PRA External** **Events**			The Combined License holder referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 PRA and Table 19.59-18 prior to fuel load. The plant specific PRA-based insight differences will be evaluated and the plant specific PRA model modified as necessary to account for the plant specific-design and, any design changes or departures from the design certification PRA.		(12) Site- and Unit-specific Conditions (f) Before initial fuel load, SCE&G shall: 7. Review differences between the as-built plant and the design used as the basis for the AP1000 probabilistic risk assessment (PRA) and the AP1000 DCD, Rev. 19, Table 19.59-18. SCE&G shall evaluate the plant-specific PRA-based insight differences and shall modify the plant-specific PRA model as necessary to account for the
21	**19.59.10-3** **Internal Fire and Internal Flood** **Analyses**			The Combined License holder referencing the AP1000 certified design will review differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analyses prior to fuel load. Plant specific internal fire and internal flood analyses will be evaluated and the analyses modified as necessary to account for the plant-specific design, and any design changes or departures from the certified		(12) Site- and Unit-specific Conditions (f) Before initial fuel load, SCE&G shall: 8. Review differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analysis. SCE&G shall evaluate the plant-specific internal fire and internal flood analyses and
22	**19.59.10-4** **Implement Severe Accident** **Management Guidance**			NOTE — addressed by proposed License Condition #6.		

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23	**19.59.10-5** **Equipment Survivability**			The Combined License holder referencing the AP1000 certified design will perform a thermal lag assessment of the as-built equipment listed in Tables 6b and 6c in Attachment A of APP-GW-GLR-069 to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. This assessment is performed prior to fuel load and is required only for equipment used for severe accident mitigation that has not been tested at severe accident conditions. The Combined License holder will assess the ability of the as-built equipment to perform during severe accident hydrogen burns using the Environment Enveloping method or the Test Based Thermal Analysis method discussed		(12) Site- and Unit-specific Conditions (f) Before initial fuel load, SCE&G shall: 9. Perform a thermal lag assessment of the as-built equipment listed in Tables 6b and 6c in Attachment A of APP-GW-GLR-069, "Equipment Survivability Assessment," to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. SCE&G shall perform this assessment for equipment used for severe accident mitigation that has not been tested at severe accident conditions. SCE&G shall assess the ability of the as-built equipment to perform
24	3. OPERATIONAL PROGRAM IMPLEMENTATION: The provisions of the regulations address implementation milestones for some operational programs. The NRC will use license conditions to ensure implementation for those operational programs whose implementation is not addressed in the regulations. COL application FSAR Subsection 13.4, Table 13.4- 201, identifies several programs required by regulations that must be implemented by a milestone to be identified in a license condition.			The licensee shall implement the programs or portions of programs identified below on or before the associated milestones identified below.	10	(10) Operational Program Implementation - SCE&G shall implement the programs or portions of programs identifiedbelow, on or before the date SCE&G achieves the following milestones.
25				The licensee shall implement each operational program identified below prior to initiating construction of nuclear safety- or security-related structures, systems, or components. None identified.		
26			18 Months Prior to Fuel Load	The license shall implement each operational program identified below at least 18 months prior to scheduled date of initial fuel load.		
27	B.1 – Reactor Operator Training		18 Months Prior to Fuel Load	B.1 – Reactor Operator Training	10.k	(k) Reactor Operator Training Program implemented 18 months before the scheduled date of initial fuel load;

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28			Receipt Of Materials	The licensee shall implement each operational program identified below prior to initial receipt of byproduct, source, or special nuclear materials onsite (excluding Exempt Quantities as described in 10 CFR 30.18).		
29	C.1 — Radiation Protection (applicable portions)		Receipt Of Materials	C.1 — Radiation Protection (applicable portions)	10.j.1	(j) Radiation Protection Program (RPP) (including the ALARA principle) or applicable portions as identified in FSAR Section 12.5 thereof: 1. RPP features applicable to receipt of by-product, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18) implemented before initial receipt of such materials;
30	C.2 – Fire Protection Program (applicable portions)		Receipt Of Materials	C.2 – Fire Protection Program (applicable portions)	10.e.1	(e) Fire Protection Program - 1. The fire protection measures in accordance with Regulatory Guide (RG) 1.189 for designated storage building areas (including adjacent fire areas that could affect the storage area) implemented before initial receipt of byproduct or special nuclear materials that are not fuel (excluding exempt quantities as described in 10 CFR 30.18);
31	C.3 – Non Licensed Plant Staff Training Program (applicable portions)		Receipt Of Materials	C.3 – Non Licensed Plant Staff Training Program (applicable portions)		
32				C.4 – Deleted		
33				C.5 – Deleted		

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34	C.6 – SNM Material Control and Accounting Program		Receipt Of Materials	C.6 – SNM Material Control and Accounting Program	10.n	(n) Special Nuclear Material Control and Accounting Program implemented before initial receipt of special nuclear material; and
35			Fuel Receipt	D. Fuel Receipt — The licensee shall implement each operational program identified below prior to initial receipt of fuel onsite:		
36	D.1 – Fire Protection (applicable portions)		Fuel Receipt	D.1 – Fire Protection (applicable portions)	10.e.2	(e) Fire Protection Program 2. The fire protection measures in accordance with RG 1.189 for areas containing new fuel (including adjacent areas where a fire could affect the new fuel) implemented before receipt of fuel onsite;
37	D.2 – Radiation Protection (applicable portions)		Fuel Receipt	D.2 – Radiation Protection (applicable portions)	10.j.2	(j) Radiation Protection Program (RPP) (including the ALARA principle) or applicable portions as identified in FSAR Section 12.5 thereof: 2. RPP features (including the ALARA principle) applicable to new fuel implemented before receipt of initial fuel on site;
38	D.3 – Special Nuclear Material Physical Protection Program		Fuel Receipt	D.3 – Special Nuclear Material Physical Protection Program	10.o	(o) Special Nuclear Material Physical Protection Program implemented before initial receipt of special nuclear material on site.
39			Fuel Receipt	D.4 – Deleted		
40			prior to initial construction testing:			
41	E.1 – Initial Test Program — Construction Testing		prior to initial construction testing:	E.1 – Initial Test Program — Construction Testing	10.m.1	(m) Initial Test Program - 1. Construction Test Program implemented before the first construction test
42			prior to initial construction testing:	E.2 — The implementation of construction and inspection procedures for steel island modules (including shield building SC modules) before and after concrete oncrete composite (SC) construction activities for seismic Category I nuclear placement, and inspection of such construction before and after concrete placement.		
43			prior to initial preoperational testing			

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44	F.1 – Initial Test Program — Preoperational Testing		prior to initial preoperational testing	F.1 – Initial Test Program — Preoperational Testing	10.m.2	(m) Initial Test Program - 2. Preoperational Test Program implemented before the first preoperational test; and
45			prior to initial fuel load	G. Fuel Loading — The licensee shall implement each operational program identified below prior to initial fuel load:		
46	G.1 – Environmental Qualification		prior to initial fuel load	G.1 – Environmental Qualification	10.a	a) Environmental Qualification Program implemented before initial fuel load;
47	G.2 – Pre-Service Testing		prior to initial fuel load	G.2 – Pre-Service Testing	10.c	(c) Preservice Testing Program implemented before initial fuel load;
48	G.3 – Process and Effluent Monitoring and Sampling		prior to initial fuel load	G.3 – Process and Effluent Monitoring and Sampling	10.f	(f) Standard Radiological Effluent Controls implemented before initial fuel load;

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49	G.4 – Radiation Protection (applicable portions)		prior to initial fuel load	G.4 – Radiation Protection (applicable portions)	10.j.3	(j) Radiation Protection Program (RPP) (including the ALARA principle) or applicable portions as identified in FSAR Section 12.5 thereof: 3. All other RPP features (including the ALARA principle) except for those applicable to control radioactive waste shipment implemented before initial fuel load;
50	G.5 – Motor-Operated Valve Testing		prior to initial fuel load	G.5 – Motor-Operated Valve Testing	10.l	(l) Motor-Operated Valve Testing Program implemented before initial fuel load;
51	G.6 – Fire Protection		prior to initial fuel load	G.6 – Fire Protection	10.e..3	(e) Fire Protection Program - 3. All fire protection
52				G.7 – Deleted		
53	G.8 – Containment Leakage Rate Testing Program		prior to initial fuel load	G.8 – Containment Leakage Rate Testing Program	10.d	(d) Containment Leakage Rate Testing Program implemented before initial fuel load;
54	G.9 – Physical Security		prior to initial fuel load	G.9 – Physical Security		
55	G.10 – Cyber Security		prior to initial fuel load	G.10 – Cyber Security		
56			prior to initial startup testing	H. Startup Testing — The licensee shall implement each operational program identified below prior to initial startup testing:		

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57	H.1 – Initial Test Program — Startup Testing		prior to initial startup testing	H.1 – Initial Test Program — Startup Testing	10.m.3	(m) Initial Test Program - 3. Startup Test Program implemented before initial fuel load;
58				I. MODE 4 – Not used		
59			prior to initial criticality	J. Initial Criticality — The licensee shall implement each operational program identified below prior to initial criticality		
60	J.1 – Reactor Vessel Material Surveillance		prior to initial criticality	J.1 – Reactor Vessel Material Surveillance	10.b	(b) Reactor Vessel Material Surveillance Program implemented before initial criticality;
61			prior to initial radioactive waste shipment	Waste Shipment — The licensee shall implement each operational program identified below prior to initial radioactive waste shipment:	10.j.4	(j) Radiation Protection Program (RPP) (including the ALARA principle) or applicable portions as identified in FSAR Section 12.5 thereof:4. RPP features (including the ALARA principle) applicable to radioactive waste shipment implemented before first shipment of radioactive waste;
62	K.1 – Radiation Protection		prior to initial radioactive waste shipment	K.1 – Radiation Protection		
63						
64						
65	A. SECURITY PROGRAM IMPLEMENTATION An implementation license condition approved in the staff requirements memo regarding SECY-05-0197 applies to the security program.		nuclear fuel is onsite (protected area), and continuing until all nuclear fuel is permanently removed from the site.	The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan and cyber security plan, and all amendments made pursuant to the authority of 10 CFR 50.90, 50.54(p), 52.97, and Section VIII of Appendix D to Part 52 when nuclear fuel is onsite (protected area), and continuing until all nuclear fuel is permanently removed from the site.		
66	B. SPECIAL NUCLEAR MATERIAL PHYSICAL PROTECTION A license condition is proposed to address when the boundary for physical protection of new fuel as SNM is required to be extended from the controlled access area (CAA) in accordance with the requirements of 10 CFR 73.67 to the operational protected area (PA) in accordance with 10 CFR 73.55.		new fuel as SNM in a controlled access area (CAA) in accordance with the requirements of 10 CFR 73.67, until such time as an operational protected area (PA)	The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan and cyber security plan, and all amendments made pursuant to the authority of 10 CFR 50.90, 50.54(p), 52.97, and Section VIII of Appendix D to Part 52 when nuclear fuel is onsite (protected area), and continuing until all nuclear fuel is permanently removed from the site.		

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67	6. OPERATIONAL PROGRAM READINESS: The NRC inspection of operational programs will be the subject of the following license condition in accordance with SECY-05-0197:			The licensee shall submit to the appropriate Director of the NRC, a schedule, no later than 12 months after issuance of the COL, that supports planning for and conduct of NRC inspections of operational programs listed in the operational program FSAR Table 13.4-201. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until either the operational programs in the FSAR table have been fully implemented or the plant has been placed in commercial service, whichever comes first. This schedule shall also address:		10. Operational Program Implementation SCE&G shall implement the programs or portions of programs identified below, on or before the date SCE&G achieves the following milestones.
68	a. the implementation of site specific Severe Accident Management Guidance. b. the reactor vessel pressurized thermal shock evaluation at least 18 months prior to initial fuel load. c. the approved preoperational and startup test procedures (including the site-specific startup administration manual (procedure) prior to initiating the plant initial test program) in accordance with FSAR Subsection 14.2.3. d. the flow accelerated corrosion (FAC) program implementation, including the construction phase activities. e. full implementation of the operational and programmatic elements of responding to an event associated with a loss of large areas of the plant due to explosions or fire, prior to initial fuel load.			a. the implementation of site specific Severe Accident Management Guidance. b. the reactor vessel pressurized thermal shock evaluation at least 18 months prior to initial fuel load. c. the approved preoperational and startup test procedures (including the site-specific startup administration manual (procedure) prior to initiating the plant initial test program) in accordance with FSAR Subsection 14.2.3. d. the flow accelerated corrosion (FAC) program implementation, including the construction phase activities. e. full implementation of the operational and programmatic elements of responding to an event associated with a loss of large areas of the plant due to explosions or fire, prior to initial fuel load.		(a) Environmental Qualification Program implemented before initial fuel load; (b) Reactor Vessel Material Surveillance Program implementedbefore initial criticality; (c) Preservice Testing Program implemented before initial fuel load; (d) Containment Leakage Rate Testing Program implemented before initial fuel load; (e) Fire Protection Program 1. The fire protection measures in accordance with Regulatory Guide (RG) 1.189 for designated storage building areas (including adjacent fire areas that could affect the storage area) implemented before initial receipt of byproduct or special nuclear materials that are not fuel (excluding exempt quantities as described in 10 CFR 30.18); 2. The fire protection measures in accordance with RG 1.189 for areas containing new fuel (including adjacent areas where a fire could affect the new fuel) implemented before receipt of fuel onsite; 3. All fire protection program features implemented before initial fuel load
69	f. the spent fuel rack Metamic coupon monitoring program implementation. g. the implementation of construction and inspection procedures for concrete filled steel plate modules activities before and after concrete placement, use of construction mock-ups, and inspection of modules before and after concrete placement as discussed in DCD Subsection 3.8.4.8. h. the availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty, prior to initial fuel load. i. the availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation, prior to initial fuel load.			f. the spent fuel rack Metamic coupon monitoring program implementation. g. the implementation of construction and inspection procedures for concrete filled steel plate modules activities before and after concrete placement, use of construction mock-ups, and inspection of modules before and after concrete placement as discussed in DCD Subsection 3.8.4.8. h. the availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty, prior to initial fuel load. i. the availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation, prior to initial fuel load.		3. All fire protection program features implemented before nitial fuel load; (f) Standard Radiological Effluent Controls implemented before initial fuel load; (g) Offsite Dose Calculation Manual implemented before initial fuel load; (h) Radiological Environmental Monitoring Program implemented before initial fuel load; (i) Process Control Program implemented before initial fuel load; (j) Radiation Protection Program (RPP) (including the ALARA principle) or applicable portions as identified in FSAR Section 12.5thereof: 1. RPP features applicable to receipt of by-product, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18) implemented before initial receipt of such materials; 2. RPP features (including the ALARA principle) applicable to new fuel implemented before receipt of initial fuel on site; 3. All other RPP features (including the ALARA principle) except for those applicable to control radioactive wasteshipment implemented before initial fuel load;

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70						4. RPP features (including the ALARA principle) applicable to radioactive waste shipment implemented before firstshipment of radioactive waste; (k) Reactor Operator Training Program implemented 18 months before the scheduled date of initial fuel load; (l) Motor-Operated Valve Testing Program implemented before initialfuel load; (m) Initial Test Program 1. Construction Test Program implemented before the first construction test; 2. Preoperational Test Program implemented before the firstpreoperational test; and 3. Startup Test Program implemented before initial fuel load; (n) Special Nuclear Material Control and Accounting Program implemented before initial receipt of special nuclear material; and (o) Special Nuclear Material Physical Protection Program implemented before initial receipt of special nuclear material on site.
71	b. the reactor vessel pressurized thermal shock evaluation at least 18			First-Plant-Only and First-Three-Plant-Only Testing A licensee shall provide written identification of the applicable references for documentation for the completion of the testing to the Director of the Office of New Reactors (or equivalent NRC management) within thirty (30) calendar days of the licensee confirmation of acceptable test results. Subsequent plant licensees crediting completion of testing by the first-plant or by the first-three-plants shall provide a report referencing the applicable documentation identified by the first (or first three) plant(s) confirming the testing to the Director of the Office of New Reactors (or equivalent NRC management). This report shall be provided to NRC either prior to initiation of pre-operational testing, or within sixty (60) days of the identification of the documentation for the completion of the testing by the first plant (or third plant, as appropriate), whichever is later.		[No Entry]
72	months prior to initial fuel load.					
73	c. the approved preoperational and startup test procedures (including					

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74	the site-specific startup administration manual (procedure) prior to					(2) Pre-operational Testing (a) SCE&G shall perform the design-specific pre-operational tests identified below: 1. In-Containment Refueling Water Storage Tank (IRWST) Heatup Test (first plant test as identified in AP1000 Design Control Document (DCD), Rev. 19, Section 14.2.9.1.3 Item (h)); 2. Pressurizer Surge Line Stratification Evaluation (first plant test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.7 Item (d)); 3. Reactor Vessel Internals Vibration Testing (first plant test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.9); 4. Core Makeup Tank Heated Recirculation Tests (first three plants test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.3 Items (k) and (w)); and 5. Automatic Depressurization System Blowdown Test (first three plants test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.3 Item (s)). (b) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(2)(a) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.9. (c) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon
75	initiating the plant initial test program) in accordance with					(3) Nuclear Fuel Loading and Pre-critical Testing (a) Until the submission of the notification required by Section 2.D.(2)(c) of this license, SCE&G shall not load fuel into the reactor vessel; (b) Upon submission of the notification required by Section 2.D.(2)(c) of this license and upon a Commission finding in accordance with 10 CFR 52.103(g) that all the acceptance criteria in the ITAAC in Appendix C to this license are met, SCE&G is authorized to perform pre-critical tests in accordance with the conditions specified herein; (c) SCE&G shall perform the pre-critical tests identified in AP1000 DCD Rev. 19, Section 14.2.10.1; (d) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(3)(c) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.10; and (e) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of the pre-critical tests identified in Section 2.D.(3)(c) of this license.

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76	FSAR Subsection 14.2.3.					(4) Initial Criticality and Low-Power Testing (a) Upon submission of the notification required by Section 2.D.(3)(e) of this license, SCE&G is authorized to operate the facility at reactor steady-state core power levels not to exceed 5-percent thermal power in accordance with the conditions specified herein; (b) SCE&G shall perform the initial criticality and low-power tests identified in AP1000 DCD Rev. 19, Sections 14.2.10.2 and 14.2.10.3, respectively, the Natural Circulation (first plant test) identified in AP1000 DCD Rev. 19, Section 14.2.10.3.6, and the Passive Residual Heat Removal Heat Exchanger (first plant test) identified in AP1000 DCD Rev. 19, Section 14.2.10.3.7; (c) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(4)(b) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.10.2 and 14.2.10.3; and (d) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of initial criticality and low-power tests identified in Section 2.D.(4)(b) of this license, including the design-specific tests identified therein.
77	d. the flow accelerated corrosion (FAC) program implementation,					(5) Power Ascension Testing (a) Upon submission of the notification required by Section 2.D.(4)(d) of this license, SCE&G is authorized to operate the facility at reactor steady-state core power levels not to exceed 100-percent thermal power in accordance with the conditions specified herein, but only for the purpose of performing power ascension testing; (b) SCE&G shall perform the power ascension tests identified in the AP1000 DCD Rev. 19, Section 14.2.10.4, the Rod Cluster Control Assembly Out of Bank Measurements (first plant test) identified in AP1000 DCD, Rev. 19, Section 14.2.10.4.6, and the Load Follow Demonstration (first plant test) identified in AP1000 DCD, Rev. 19, Section 14.2.10.4.22; (c) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(5)(b) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev.19, Section 14.2.10.4; and (d) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of power ascension tests identified in Section 2.D.(5)(b) of this license, including the design-specific tests identified therein. (6) Maximum Power Level Upon submission of the notification required by Section 2.D.(5)(d) of this license, SCE&G is authorized to operate the facility at steady state

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78	including the construction phase activities.			The issuance of this COL, subject to the Environmental Protection Plan and the conditions for the protection of the environment set forth herein, is in accordance with the National Environmental Policy Act of 1969, as amended, and with applicable sections of 10 CFR 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," as referenced by Subpart C of 10 CFR 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," and all applicable requirements therein have been satisfied.		(8) The Technical Specifications, Environmental Protection Plan, and ITAAC in Appendices A, B, and C, respectively of this license, as revised through Amendment No. 22, are hereby ncorporated into this license.
79	**EMERGENCY PLANNING ACTIONS:**		at least 180 days prior to initial fuel load		12.c	(c) No later than 180 days before initial fuel load, SCE&G shall submit to the Director of NRO, or the Director's designee, in writing:
80	**EMERGENCY PLANNING ACTIONS:**		at least 180 days prior to initial fuel load	The licensee shall submit a fully developed set of plant-specific Emergency Action Levels (EALs) for VCSNS Units 2 and 3 in accordance with NEI-07-01 Revision 0. These fully developed EALs shall be submitted to the NRC for confirmation at least 180 days prior to initial fuel load. The submitted EALs will be written with no deviations	12.c.1	1. A fully developed set of plant-specific emergency action levels (EALs) for VCSNS Unit 2 in accordance with Nuclear Energy Institute (NEI) 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, with no deviations. The EALs shall have been discussed and agreed upon with State and local officials
81			at least 180 days prior to initial fuel load		12.c.2	2. An assessment of emergency response staffing performed in accordance with NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0.
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83	of responding to an event associated with a loss of large areas of					
84	the plant due to explosions or fire, prior to initial fuel load.			There is no license condition for Fukishima Actions ifor Mitigating Strategies n the VC Summer COL Application		There are no Fukishima Actions for Mitigating Strategies in the the VC Summer License
85				There is no license condition for Fukishima Actions for Reliable Spent Fuel Pool Level Instrumentation in the VC Summer COL Application		There are no Fukishima Actions for Reliable Spent Fuel Pool Level Instrumentation in the the VC Summer License

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86	N/A			There is no license condition for Fukishima Actions for Emergency Planning Actions in the VC Summer COL Application		There are no Fukishima Actions for Emergency Planning Actions in the the VC Summer License
87				[No Entry]		[No Entry]

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67	The FPL COLA has the following two additional items are listed for operational programs in the PTN COLA that the VCS COLA does not have. However, both of these are listed as ITAAC in Appendix C of the VC Summer License. The license also includes an item (12) Site- and Unit-specific Conditions which includes many of the items in column J.
68	a. the emergency planning implementation procedures to the NRC consistent with 10 CFR Part 50, Appendix E, Section V. e. an emergency response data system (ERDS) implementation program plan consistent with 10 CFR Part 50, Appendix E, Section V. It appears that the NR C included all the programs from Chapter 14 and included them as a license condition.
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71	PTN COLA contains reporting requirements not explicitly included in the VCSNS COL.
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78	No difference.
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84	Since OL's and COL's received NRC Orders for Fukishima actions, it is expected that there would not be License Conditions for Fukishima Actions for Mitigating Strategies for VC Summer.
85	Since OL's and COL's received NRC Orders for Fukishima actions, it is expected that there would not be License Conditions for Fukishima Actions for Reliable Spent Fuel Pool Level Instrumentation for VC Summer.

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86	Since OL's and COL's received NRC Orders for Fukishima actions, it is expected that there would not be License Conditions for Fukishima Actions for Emergency Planning Actions for VC Summer.
87	PTN COLA contains a proposed LC not listed in the VCSNS COL.

COMBINED LICENSE

VIRGIL C. SUMMER NUCLEAR STATION UNIT 2

SOUTH CAROLINA ELECTRIC AND GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

Docket No. 52-027

License No. NPF-93

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for a combined license (COL) for Virgil C. Summer Nuclear Station (VCSNS) Unit 2 filed by South Carolina Electric & Gas Company (SCE&G), acting on behalf of itself and South Carolina Public Service Authority (Santee Cooper), herein referred to as "the VCSNS owners," which incorporates by reference Appendix D to 10 CFR Part 52, complies with the applicable standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. There is reasonable assurance that the facility will be constructed and will operate in conformity with the application, as amended, the provisions of the Act, and the Commission regulations set forth in 10 CFR Chapter I, except as exempted from compliance in Section 2.F below;

   C. There is reasonable assurance (i) that the activities authorized by this COL can be conducted without endangering the health and safety of the public and (ii) that such activities will be conducted in compliance with the Commission regulations set forth in 10 CFR Chapter I, except as exempted from compliance in Section 2.F below;

   D. SCE&G[1] is technically qualified to engage in the activities authorized by this license in accordance with the Commission regulations set forth in 10 CFR Chapter I. The VCSNS owners are financially qualified to engage in the activities authorized by this COL in accordance with the Commission regulations set forth in 10 CFR Chapter I;

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[1] SCE&G is authorized by Santee Cooper to exercise responsibility and control over the physical construction, operation, and maintenance of the facility.

E. The VCSNS owners have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements;"

F. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

G. After weighing the environmental, economic, technical, and other benefits of the facility against environmental and other costs and considering reasonable available alternatives, the issuance of this license subject to the conditions for protection of the environment set forth herein is in accordance with Subpart A of 10 CFR Part 51 and all applicable requirements have been satisfied; and

H. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with the applicable regulations in 10 CFR Parts 30, 40, and 70.

2. On the basis of the foregoing findings regarding this facility, COL No. NPF-93 is hereby issued to SCE&G and Santee Cooper (the licensees), to read as follows:

A. This license applies to the VCSNS Unit 2, a light-water nuclear reactor and associated equipment (the facility), owned by the VCSNS owners. The facility would be located approximately 1 mile from the center of VCSNS Unit 1 in western Fairfield County, approximately 15 miles west of Winnsboro, and 26 miles northwest of Columbia, SC and is described in the licensee's final safety analysis report (FSAR), as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) (a) SCE&G, pursuant to Sections 103 and 185b. of the Act and 10 CFR Part 52, to construct, possess, use, and operate the facility at the designated location in accordance with the procedures and limitations set forth in this license;

(b) Santee Cooper pursuant to the Act and 10 CFR Part 52, to possess but not operate the facility at the designated location in Fairfield County, South Carolina, in accordance with the procedures and limitations set forth in this license;

(2) (a) SCE&G, pursuant to the Act and 10 CFR Part 70, to receive and possess at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and in amounts necessary for reactor operation, described in the FSAR, as supplemented and amended;

(b) SCE&G, pursuant to the Act and 10 CFR Part 70, to use special nuclear material as reactor fuel, after a Commission finding under 10 CFR 52.103(g) has been made, in accordance with the limitations for storage and in amounts necessary for reactor operation, described in the FSAR, as supplemented and amended;

(3) (a) SCE&G, pursuant to the Act and 10 CFR Parts 30 and 70, to receive, possess, and use, at any time before a Commission finding under 10 CFR 52.103(g), such byproduct and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts, as necessary;

(b) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, after a Commission finding under 10 CFR 52.103(g), any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as necessary;

(4) (a) SCE&G, pursuant to the Act and 10 CFR Parts 30 and 70, to receive, possess, and use, before a Commission finding under 10 CFR 52.103(g), in amounts not exceeding those specified in 10 CFR 30.72, any byproduct or special nuclear material that is (1) in unsealed form; (2) on foils or plated surfaces, or (3) sealed in glass, for sample analysis or instrument calibration or other activity associated with radioactive apparatus or components;

(b) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use, after a Commission finding under 10 CFR 52.103(g), in amounts as necessary, any byproduct, source, or special nuclear material without restriction as to chemical or physical form, for sample analysis or instrument calibration or other activity associated with radioactive apparatus or components but not uranium hexafluoride; and

(5) SCE&G, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. The license is subject to, and the licensees shall comply with, all applicable provisions of the Act and the rules, regulations, and orders of the Commission, including the conditions set forth in 10 CFR Chapter I, now or hereafter in effect.

D. The license is subject to, and SCE&G shall comply with, the conditions specified and incorporated below:

(1) <u>Changes during Construction</u>

(a) SCE&G may request use of a preliminary amendment request (PAR) process, for license amendments, at any time before a Commission finding under 10 CFR 52.103(g). To use the PAR process, SCE&G shall submit a written request to the Office of New Reactors (NRO) in accordance with COL-ISG-025, “Changes during Construction under Part 52.”

(b) Before NRO's issuance of a written PAR notification, SCE&G shall submit the license amendment request (LAR). Thereafter, NRO will issue a written PAR notification, setting forth whether SCE&G may proceed in accordance with the PAR, LAR, and COL-ISG-025. If SCE&G elects to proceed and the LAR is subsequently denied, SCE&G shall return the facility to its current licensing basis.

(2) <u>Pre-operational Testing</u>

(a) SCE&G shall perform the design-specific pre-operational tests identified below:

1. In-Containment Refueling Water Storage Tank (IRWST) Heatup Test (first plant test as identified in AP1000 Design Control Document (DCD), Rev. 19, Section 14.2.9.1.3 Item (h));

2. Pressurizer Surge Line Stratification Evaluation (first plant test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.7 Item (d));

3. Reactor Vessel Internals Vibration Testing (first plant test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.9);

4. Core Makeup Tank Heated Recirculation Tests (first three plants test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.3 Items (k) and (w)); and

5. Automatic Depressurization System Blowdown Test (first three plants test as identified in AP1000 DCD, Rev. 19, Section 14.2.9.1.3 Item (s)).

(b) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(2)(a) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.9.

(c) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of the design-specific pre-operational tests identified in Section 2.D.(2)(a) of this license; and

(d) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon the successful completion of all the ITAAC included in Appendix C to this license.

(3) Nuclear Fuel Loading and Pre-critical Testing

(a) Until the submission of the notification required by Section 2.D.(2)(c) of this license, SCE&G shall not load fuel into the reactor vessel;

(b) Upon submission of the notification required by Section 2.D.(2)(c) of this license and upon a Commission finding in accordance with 10 CFR 52.103(g) that all the acceptance criteria in the ITAAC in Appendix C to this license are met, SCE&G is authorized to perform pre-critical tests in accordance with the conditions specified herein;

(c) SCE&G shall perform the pre-critical tests identified in AP1000 DCD Rev. 19, Section 14.2.10.1;

(d) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(3)(c) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.10; and

(e) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of the pre-critical tests identified in Section 2.D.(3)(c) of this license.

(4) Initial Criticality and Low-Power Testing

(a) Upon submission of the notification required by Section 2.D.(3)(e) of this license, SCE&G is authorized to operate the facility at reactor steady-state core power levels not to exceed 5-percent thermal power in accordance with the conditions specified herein;

(b) SCE&G shall perform the initial criticality and low-power tests identified in AP1000 DCD Rev. 19, Sections 14.2.10.2 and 14.2.10.3, respectively, the Natural Circulation (first plant test) identified in AP1000 DCD Rev. 19, Section 14.2.10.3.6, and the Passive Residual Heat Removal Heat Exchanger (first plant test) identified in AP1000 DCD Rev. 19, Section 14.2.10.3.7;

(c) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(4)(b) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev. 19, Section 14.2.10.2 and 14.2.10.3; and

(d) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of initial criticality and low-power tests identified in Section 2.D.(4)(b) of this license, including the design-specific tests identified therein.

(5) Power Ascension Testing

(a) Upon submission of the notification required by Section 2.D.(4)(d) of this license, SCE&G is authorized to operate the facility at reactor steady-state core power levels not to exceed 100-percent thermal power in accordance with the conditions specified herein, but only for the purpose of performing power ascension testing;

(b) SCE&G shall perform the power ascension tests identified in the AP1000 DCD Rev. 19, Section 14.2.10.4, the Rod Cluster Control Assembly Out of Bank Measurements (first plant test) identified in AP1000 DCD, Rev. 19, Section 14.2.10.4.6, and the Load Follow Demonstration (first plant test) identified in AP1000 DCD, Rev. 19, Section 14.2.10.4.22;

(c) SCE&G shall review and evaluate the results of the tests identified in Section 2.D.(5)(b) of this license and confirm that these test results are within the range of acceptable values predicted or otherwise confirm that the tested systems perform their specified functions in accordance with AP1000 DCD Rev.19, Section 14.2.10.4; and

(d) SCE&G shall notify the Director of NRO, or the Director's designee, in writing, upon successful completion of power ascension tests identified in Section 2.D.(5)(b) of this license, including the design-specific tests identified therein.

(6) Maximum Power Level

Upon submission of the notification required by Section 2.D.(5)(d) of this license, SCE&G is authorized to operate the facility at steady state reactor core power levels not to exceed 3400 MW thermal (100-percent thermal power), as described in the FSAR, in accordance with the conditions specified herein.

(7) Reporting Requirements

(a) Within 30 days of a change to the initial test program described in FSAR Section 14, Initial Test Program, made in accordance with 10 CFR 50.59 or in accordance with 10 CFR Part 52, Appendix D, Section VIII, "Processes for Changes and Departures," SCE&G shall report the change to the Director of NRO, or the Director's designee, in accordance with 10 CFR 50.59(d).

(b) SCE&G shall report any violation of a requirement in Section 2.D.(3), Section 2.D.(4), Section 2.D.(5), and Section 2.D.(6) of this license within 24 hours. Initial notification shall be made to the NRC Operations Center in accordance with 10 CFR 50.72, with written follow up in accordance with 10 CFR 50.73.

(8) Incorporation

The Technical Specifications, Environmental Protection Plan, and ITAAC in Appendices A, B, and C, respectively of this license, as revised through Amendment No. 22, are hereby incorporated into this license.

(9) Technical Specifications

The technical specifications in Appendix A to this license become effective upon a Commission finding that the acceptance criteria in this license (ITAAC) are met in accordance with 10 CFR 52.103(g).

(10) Operational Program Implementation

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SCE&G shall implement the programs or portions of programs identified below, on or before the date SCE&G achieves the following milestones.

(a) Environmental Qualification Program implemented before initial fuel load;

(b) Reactor Vessel Material Surveillance Program implemented before initial criticality;

(c) Preservice Testing Program implemented before initial fuel load;

(d) Containment Leakage Rate Testing Program implemented before initial fuel load;

(e) Fire Protection Program

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1. The fire protection measures in accordance with Regulatory Guide (RG) 1.189 for designated storage building areas (including adjacent fire areas that could affect the storage area) implemented before initial receipt of byproduct or special nuclear materials that are not fuel (excluding exempt quantities as described in 10 CFR 30.18);

2. The fire protection measures in accordance with RG 1.189 for areas containing new fuel (including adjacent areas where a fire could affect the new fuel) implemented before receipt of fuel onsite;

3. All fire protection program features implemented before initial fuel load;

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(f) Standard Radiological Effluent Controls implemented before initial fuel load;

These 2 are combined in PTN

(g) Offsite Dose Calculation Manual implemented before initial fuel load;

(h) Radiological Environmental Monitoring Program implemented before initial fuel load;

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(i) Process Control Program implemented before initial fuel load;

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(j) Radiation Protection Program (RPP) (including the ALARA principle) or applicable portions as identified in FSAR Section 12.5 thereof:

1. RPP features applicable to receipt of by-product, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18) implemented before initial receipt of such materials;

2. RPP features (including the ALARA principle) applicable to new fuel implemented before receipt of initial fuel on site;

3. All other RPP features (including the ALARA principle) except for those applicable to control radioactive waste shipment implemented before initial fuel load;

4. RPP features (including the ALARA principle) applicable to radioactive waste shipment implemented before first shipment of radioactive waste;

(k) Reactor Operator Training Program implemented 18 months before the scheduled date of initial fuel load;

(l) Motor-Operated Valve Testing Program implemented before initial fuel load;

(m) Initial Test Program

1. Construction Test Program implemented before the first construction test;

2. Preoperational Test Program implemented before the first preoperational test; and

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3. Startup Test Program implemented before initial fuel load;

(n) Special Nuclear Material Control and Accounting Program implemented before initial receipt of special nuclear material; and

(o) Special Nuclear Material Physical Protection Program implemented before initial receipt of special nuclear material on site.

(11) Operational Program Implementation Schedule

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No later than 12 months after issuance of the COL, SCE&G shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the operational programs listed in FSAR Table 13.4-201, including the associated estimated date for initial loading of fuel. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until all the operational programs listed in FSAR Table 13.4-201 have been fully implemented.

(12) Site- and Unit-specific Conditions

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(a) Before commencing installation of individual piping segments and connected components in their final locations, SCE&G shall complete the as-designed pipe rupture hazards analysis for compartments (rooms) containing those segments in accordance with the criteria outlined in the AP1000 DCD, Rev. 19, Sections 3.6.1.3.2 and 3.6.2.5, and shall inform the Director of NRO, or the Director's designee, in writing, upon the completion of this analysis and the availability of the as-designed pipe rupture hazards analysis reports.

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(b) Before commencing installation of individual piping segments identified in AP1000 DCD, Rev. 19, Section 3.9.8.7, and connected components in their final locations in the facility, SCE&G shall complete the analysis of the as-designed individual piping segments and shall inform the Director of NRO, or the Director's designee, in writing, upon the completion of these analyses and the availability of the design reports for the selected piping packages.

(c) No later than 180 days before initial fuel load, SCE&G shall submit to the Director of NRO, or the Director's designee, in writing:

1. A fully developed set of plant-specific emergency action levels (EALs) for VCSNS Unit 2 in accordance with Nuclear Energy Institute (NEI) 07-01, "Methodology for Development of Emergency Action Levels Advanced Passive Light Water Reactors," Revision 0, with no deviations. The EALs shall have been discussed and agreed upon with State and local officials.

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2. An assessment of emergency response staffing performed in accordance with NEI 10-05, "Assessment of On-Shift Emergency Response Organization Staffing and Capabilities," Revision 0.

(d) SCE&G shall not revise or modify the provisions of Sections 5.3, 5.4, 5.6, 5.9, and 5.10 of the Special Nuclear Material (SNM) Physical Protection Program until the requirements of 10 CFR 73.55 are implemented.

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(e) No later than 12 months after issuance of the COL, SCE&G shall submit to the Director of NRO, or the Director's designee, a schedule for implementation of the following license conditions. The schedule shall be updated every 6 months until 12 months before scheduled fuel loading, and every month thereafter until each license condition has been fully implemented. The schedule shall identify the completion of or implementation of the following:

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1. The construction and inspection procedures for steel concrete composite (SC) construction activities for seismic Category I nuclear island modules (including shield building SC modules) described in AP1000 DCD Rev. 19, Section 3.8.4.8;

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2. The spent fuel rack Metamic Coupon monitoring program (before initial fuel load);

PTN LC not word for word, PTN does not have "(before initial fuel load)"

3. Implementation of the flow accelerated corrosion (FAC) program including construction phase activities (before initial fuel load);

4. A turbine maintenance and inspection program, which must be consistent with the maintenance and inspection program plan activities and inspection intervals identified in FSAR Section 10.2.3.6 (before initial fuel load);

5. The availability of documented instrumentation uncertainties to calculate a power calorimetric uncertainty (before initial fuel load);

6. The availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation (before initial fuel load);

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7. The site-specific severe accident management guidelines (before startup testing);

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8. The operational and programmatic elements of the mitigative strategies for responding to circumstances associated with loss of large areas of the plant due to explosions or fire developed in accordance with 10 CFR 50.54(hh)(2) (before initial fuel load); and

9. The pre-operational and startup procedures (including the site-specific startup administration manual) identified in FSAR Section 14.2.3 (before initiating the initial test program).

(f) Before initial fuel load, SCE&G shall:

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1. Update the seismic interaction analysis in AP1000 DCD, Rev. 19, Section 3.7.3.5 to reflect as-built information, which must be based on as-procured data, as well as the as-constructed condition;

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2. Reconcile the seismic analyses described in Section 3.7.2 of the AP1000 DCD, Rev. 19, to account for detailed design changes, including, but not limited to, those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information;

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3. Calculate the instrumentation uncertainties of the actual plant operating instrumentation to confirm that either the design limit departure from nucleate boiling ratio (DNBR) values remain valid or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties;

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4. Update the pressure-temperature (P-T) limits using the pressure temperature limits report (PTLR) methodologies approved in AP1000 DCD, Rev. 19, using the plant-specific material properties or confirm that the reactor vessel material properties meet the specifications of and use the Westinghouse generic PTLR curves;

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5. Verify that plant-specific belt line material properties are consistent with the properties given in AP1000 DCD Rev. 19, Section 5.3.3.1 and Tables 5.3-1 and 5.3-3. The verification must include a pressurized thermal shock (PTS) evaluation based on as-procured reactor vessel material data and the projected neutron fluence for the plant design objective. Submit this PTS evaluation report to the Director of NRO, or the Director's designee, in writing, at least 18 months before initial fuel load;

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6. Review differences between the as-built plant and the design used as the basis for the AP1000 seismic margin analysis. SCE&G shall perform a verification walkdown to identify differences between the as-built plant and the design. SCE&G shall evaluate any differences and must modify the seismic margin analysis as necessary to account for the plant-specific design and any design

changes or departures from the certified design. SCE&G shall compare the as-built structures, systems, and components (SSC) high confidence, low probability of failures (HCLPFs) with those assumed in the AP1000 seismic margin evaluation, before initial fuel load. SCE&G shall evaluate deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis to determine if vulnerabilities have been introduced;

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7. Review differences between the as-built plant and the design used as the basis for the AP1000 probabilistic risk assessment (PRA) and the AP1000 DCD, Rev. 19, Table 19.59-18. SCE&G shall evaluate the plant-specific PRA-based insight differences and shall modify the plant-specific PRA model as necessary to account for the plant-specific design and any design changes or departure from the PRA certified in Rev. 19 of the AP1000 DCD;

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8. Review differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analysis. SCE&G shall evaluate the plant-specific internal fire and internal flood analyses and shall modify the analyses as necessary to account for the plant-specific design and any design changes or departures from the design certified in Rev. 19 of the AP1000 DCD; and

PTN LC not word for word

9. Perform a thermal lag assessment of the as-built equipment listed in Tables 6b and 6c in Attachment A of APP-GW-GLR-069, "Equipment Survivability Assessment," to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. SCE&G shall perform this assessment for equipment used for severe accident mitigation that has not been tested at severe accident conditions. SCE&G shall assess the ability of the as-built equipment to perform during accident hydrogen burns using the environment enveloping method or the test based thermal analysis method described in Electric Power Research Institute (EPRI) NP-4354, "Large Scale Hydrogen Burn Equipment Experiments."

10. Implement a surveillance program for explosively actuated valves (squib valves) that includes the following provisions in addition to the requirements specified in the edition of the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) as incorporated by reference in 10 CFR 50.55a.

a. Preservice Testing

All explosively actuated valves shall be preservice tested by verifying the operational readiness of the actuation logic and associated electrical circuits for each explosively actuated valve with its pyrotechnic charge removed from the valve. This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available at the explosively actuated valve from each circuit that is relied upon to actuate the valve. In addition, a sample of at least 20% of the pyrotechnic charges in all explosively actuated valves shall be tested in the valve or a qualified test fixture to confirm the capability of each sampled pyrotechnic charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping. The sampling must select at least one explosively actuated valve from each redundant safety train. Corrective action shall be taken to resolve any deficiencies identified in the operational readiness of the actuation logic or associated electrical circuits, or the capability of a pyrotechnic charge. If a charge fails to fire or its capability is not confirmed, all charges with the same batch number shall be removed, discarded, and replaced with charges from a different batch number that has demonstrated successful 20% sampling of the charges.

b. Operational Surveillance

Explosively actuated valves shall be subject to the following surveillance activities after commencing plant operation:

i. At least once every 2 years, each explosively actuated valve shall undergo visual external examination and remote internal examination (including evaluation and removal of fluids or contaminants that may interfere with operation of the valve) to verify the operational readiness of the valve and its actuator. This examination shall also verify the appropriate position of the internal actuating mechanism and proper operation of remote position indicators. Corrective action shall be taken to resolve any deficiencies identified during the examination with post-maintenance testing conducted that satisfies the preservice testing requirements.

ii. At least once every 10 years, each explosively actuated valve shall be disassembled for internal examination of the valve and actuator to verify the operational readiness of the valve assembly and the integrity of individual components and to remove any foreign material, fluid, or corrosion. The examination schedule shall provide for both of the two valve designs used for explosively actuated valves at the facility to be included among the explosively actuated valves to be disassembled and examined every 2 years. Corrective action shall be taken to resolve any deficiencies identified during the examination with post-maintenance testing conducted that satisfies the preservice testing requirements.

iii. For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the operational readiness of the actuation logic and associated electrical circuits shall be verified for each sampled explosively actuated valve following removal of its charge. This must include confirmation that sufficient electrical parameters (voltage, current, resistance) are available for each valve actuation circuit. Corrective action shall be taken to resolve any deficiencies identified in the actuation logic or associated electrical circuits.

iv. For explosively actuated valves selected for test sampling every 2 years in accordance with the ASME OM Code, the sampling must select at least one explosively actuated valve from each redundant safety train. Each sampled pyrotechnic charge shall be tested in the valve or a qualified test fixture to confirm the capability of the charge to provide the necessary motive force to operate the valve to perform its intended function without damage to the valve body or connected piping. Corrective action shall be taken to resolve any deficiencies identified in the capability of a pyrotechnic charge in accordance with the preservice testing requirements.

This license condition shall expire upon (1) incorporation of the above surveillance provisions for explosively actuated valves into the facility's inservice testing program, or (2) incorporation of inservice testing requirements for explosively actuated valves in new reactors (i.e., plants receiving a construction permit, or combined license for construction and operation, after January 1, 2000) to be specified in a future edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a, including any conditions imposed by the NRC, into the facility's inservice testing program.

(13) Mitigation Strategies for Beyond-Design Basis External Events

SCE&G shall address the following requirements:

These LC's are in PTNPart 10 under Fukishima Actions

(a) SCE&G shall develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment and spent fuel pool cooling capabilities following a beyond-design-basis external event.

(b) These strategies must be capable of mitigating a simultaneous loss of all AC power and loss of normal access to the normal heat sink and have adequate capacity to address challenges to core cooling, containment, and spent fuel pool cooling capabilities at all units on the VCSNS site.

(c) SCE&G must provide reasonable protection for the associated equipment from external events. Such protection must demonstrate that there is adequate capacity to address challenges to core cooling, containment, and spent fuel pool cooling capabilities at all units on the VCSNS site.

(d) SCE&G must be capable of implementing the strategies in all modes.

(e) Full compliance shall include procedures, guidance, training, and acquisition, staging, or installing of equipment needed for the strategies.

(f) SCE&G shall promptly start implementation of the requirements stated in this condition and shall complete full implementation prior to initial fuel load.

1. SCE&G shall, within twenty (20) days of issuance of this license, notify the Commission (1) if they are unable to comply with any of these requirements, (2) if compliance with any of the requirements is unnecessary in their

specific circumstances, or (3) if implementation of any of the requirements would cause SCE&G to be in violation of provisions of any Commission regulation or license. The notification shall provide SCE&G's justification for seeking relief from or variation of any specific requirement.

2. If SCE&G considers that implementation of any of these requirements would adversely impact safe and secure operation of the facility, SCE&G must notify the Commission, within twenty (20) days of issuance of the license, of the adverse safety impact, the basis for their determination that the requirement has an adverse safety impact, and either a proposal for achieving the same objectives specified in this license condition, or a schedule for modifying the facility to address the adverse safety condition. If neither approach is appropriate, then SCE&G must supplement their response to paragraph 2.D.(13)(f)1. of this license to identify the condition as a requirement with which they cannot comply, with attendant justifications as required in paragraph 2.D.(13)(f)1. of this license.

3. SCE&G shall, within one (1) year after issuance of the NRC's final Interim Staff Guidance detailing an acceptable approach for complying with these requirements, submit to the Commission for review an overall integrated plan, including a description of how compliance with the requirements described in section 2.D.(13) of this license will be achieved.

4. SCE&G shall provide an initial status report sixty (60) days following issuance of the final Interim Staff Guidance and at six (6)-month intervals following submittal of the overall integrated plan, as required in paragraph 2.D.(13)(f)3. of this license, which delineates progress made in implementing the requirements of this license condition

5. SCE&G shall report to the Commission when full compliance with the requirements described in section 2.D.(13) of this license is achieved.

6. SCE&G responses to conditions 2.D.(13)(f)1., 2.D.(13)(f)2., 2.D.(13)(f)3., 2.D.(13)(f)4., and 2.D.(13)(f)5. of this license, shall be submitted in accordance with 10 CFR 52.3.

E. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

F. Exemptions

(1) The following exemptions from any part of the referenced design certification rule meet the requirements of 10 CFR 52.7 and Section VIII.A.4, VIII.B.4, or VIII.C.4 of Appendix D to 10 CFR Part 52, are authorized by law, will not present an undue risk to the public health or safety, and are consistent with the common defense and security. Special circumstances are present in that the application of the regulation in these particular circumstance are not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)) as described in the application and the staff SER dated August 17, 2011. In addition, for exemption 2.F.(1)(b) the exemption will not result in a significant decrease in the level of safety otherwise provided by the design, and the special circumstances outweigh any decrease in the safety that may result from the reduction in standardization caused by the exemption.

(a) The licensees are exempt from the requirement of 10 CFR Part 52, Appendix D, Section IV.A.2.a to include a plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for the AP1000 certified design. This exemption is specific to the organization and numbering scheme in the FSAR and is related to departure number STD DEP 1.1-1 and VCS DEP 2.0-1.

(b) The licensees are exempt from the requirement of 10 CFR Part 52, Appendix D, Section IV.A.2d to include information demonstrating compliance with the site parameters and interface requirements. This exemption is specific to the maximum safety wet bulb (noncoincident) air temperature in the FSAR and is related to departure number VCS DEP 2.0-2.

(2) The following exemptions from regulations were granted in the rulemaking for the design certification rule that is referenced in the application. In accordance with 10 CFR Part 52, Appendix D, Section V, Applicable Regulations, Subsection B, and pursuant to 10 CFR 52.63(a)(5), the licensees are exempt from portions of the following regulations:

(a) Paragraph (f)(2)(iv) of 10 CFR 50.34—Plant Safety Parameter Display Console;

(b) Paragraph (c)(1) of 10 CFR 50.62—Auxiliary (or emergency) feedwater system; and

(c) Appendix A to 10 CFR Part 50, GDC 17—Second offsite power supply circuit.

(3) For the reasons set forth below, the following specific exemptions which are outside the scope of the design certification rule referenced in the application are granted:

(a) The licensees are exempt from the requirements of 10 CFR 70.22(b), 10 CFR 70.32(c), 10 CFR 74.31, 10 CFR 74.41, and 10 CFR 74.51 because the licensees meet the requirements of 10 CFR 70.17 and 74.7 as follows. The exemption is authorized by law, will not present an undue risk to the public health or safety, and is consistent with the common defense and security. Additionally, special circumstances are present in that the application of the regulations in this particular circumstance is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)) as described in the FSAR and the staff SER dated August 17, 2011.

(b) The licensees are exempt from the requirements of 10 CFR 52.93(a)(1) as it relates to the exemption granted in Section 2.F.(1)(a) of this license because the exemption meets the requirements of 10 CFR 52.7, because the exemption is authorized by law, will not present an undue risk to the public health or safety, and is consistent with the common defense and security. Additionally, special circumstances are present in that the application of the regulation in this particular circumstance is not necessary to achieve the underlying purpose of the rule (10 CFR 50.12(a)(2)(ii)) as described in the staff SER dated August 17, 2011.

G. Following SCE&G's ITAAC closure notifications under paragraph (c)(1) of 10 CFR 52.99 until the Commission makes the finding under 10 CFR 52.103(g), SCE&G shall notify the NRC, in a timely manner, of new information that materially alters the bases for determining that either inspections, tests, or analyses were performed as required, or that acceptance criteria are met. The notification must contain sufficient information to demonstrate that, notwithstanding the new information, the prescribed inspections, tests, or analyses have been performed as required, and the prescribed acceptance criteria are met.

H. SCE&G shall maintain the guidance and strategies developed in accordance with 10 CFR 50.54(hh)(2).

I. This license is effective as of March 30, 2012 and shall expire at midnight on the date 40 years from the date that the Commission finds that the acceptance criteria in the combined license are met in accordance with 10 CFR 52.103(g).

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Michael R. Johnson, Director
Office of New Reactors

Appendices:
Appendix A – Technical Specifications
Appendix B – Environmental Protection Plan
Appendix C – Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)