UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	) **Docket Nos. 52-025-COL and 52-026-COL**
Southern Nuclear Operating Company	)
(COL Application for Vogtle Electric Generating Plant, Units 3 and 4)	) **October 17, 2011**

SOUTHERN NUCLEAR OPERATING COMPANY'S RESPONSE TO THE COMMISSION'S ORDER OF OCTOBER 6, 2011

On October 6, 2011, the Nuclear Regulatory Commission ("Commission" or "NRC") issued the "Order (Supplemental Responses and Post-Hearing Questions)" providing a listing of questions for which written follow-up was requested or offered during the mandatory hearing for the Vogtle Units 3 and 4 combined license application ("COLA")[1] and associated request for a Limited Work Authorization ("LWA-B").[2] Pursuant to the Order, Southern Nuclear Operating Company ("SNC") hereby responds to the Commission's questions. SNC provides written follow-up to Items G and L, as well as answers to Questions 1, 2, 7, 8, 9, 12, 13 and 14.

As noted in SNC's Response to the Commission's Order of August 31, 2011 (filed September 13, 2011), to the extent practicable and consistent with providing full and complete answers to the Commission's questions, SNC has attempted to include information in its

[1] Acronyms not defined herein are those provided in the Order, at note 1.

[2] Order (Supplemental Responses and Post-Hearing Questions), Docket Nos. 52-025-COL & 52-026-COL (Oct. 6, 2011) ("Order").

responses that is within the scope of the mandatory hearing as described in the relevant Staff Requirements Memoranda.[3]

RESPONSE TO MANDATORY HEARING ITEMS

Item	Date	Panel	Transcript page(s) and line number(s)
G	Sept. 27, 2011	Safety Panel 3	p. 231, lines 8-21

Response: An initial response to this question was provided in the closing testimony of Joseph A. Miller and is recorded at page 348 of the Transcript.[4] The statement by Mr. Miller is as follows:

> Relative to seismic questions, a question came about the meaning of [F]S[A]R Section 19.55.6.3 during the discussion of seismic margins. We have reviewed the wording in that section and while we recognize the wording could be crafted in a more artful fashion, the information provided is correct. The section means that for site specific conditions concerning soil related failure modes, the demonstration of adequate seismic margin is performed for a review level earthquake equal to 1.67 times the Vogtle GMRS. Thus, both the calculated seismic -- specifics seismic response and the seismic loads are scaled up by a factor of 1.67. We can answer further questions about that if desired.

Further expanding on Mr. Miller's statements, the two phrases in the relevant statement from FSAR Section 19.55.6.3 address two separate, but related, issues.

- "For site specific conditions relating to soil-related failure modes, the demonstration of adequate seismic margin of the AP1000 design at the [Vogtle Electric Generating Plant (VEGP)] site is performed for a review level earthquake of 1.67 x VEGP GMRS" defines the ground motion for the seismic margin assessment.
- The second reference to "where the VEGP site-specific review level earthquake seismic responses and seismic loads are defined as 1.67 x VEGP GMRS seismic responses and seismic loads" refers to the assumption used to predict the seismic responses and loads resulting from the review level earthquake.

[3] *See* SECY-11-0042, Revisions To Internal Commission Procedures Section On Mandatory Hearings (Mar. 25, 2011); SECY-10-0082, Mandatory Hearing Process For Combined License Application Proceedings Under 10 C.F.R. Part 52 (Dec. 23, 2010).

[4] Consistent with the page numbers used in the Order, all page numbers referencing the Mandatory Hearing transcript refer to the uncorrected Transcript appended to the Order. SNC notes, however, that on October 11, 2011, the NRC Staff filed a joint motion on behalf of Staff and SNC containing transcript corrections. Where SNC has corrected transcript language in accordance with that motion herein, the changed language appears in brackets.

Thus, the dual references to "1.67 times the GMRS" merely acknowledge the conservative assumption that the seismic event and the resulting seismic responses and loads increase by the same factor. The assumption that greater ground motion would produce equivalently greater seismic responses and loads is conservative because seismic responses and loads generally increase at a lower rate than ground motion.

Item	Date	Panel	Transcript page(s) and line numbers(s)
L	Sept. 28, 2011	Environmental Panel 1	p. 325, line 24 (identifying speaker), line 25 (last word, start of question); p. 326, lines 1-13

Response: The information requested under this Item is provided in response to Question 14 below.

RESPONSES TO POST-HEARING QUESTIONS

No.	Category	Reference	Directed To	Question
1	Safety	General	Staff and Applicant	In the event the Commission decides to impose a license condition requiring implementation of all Commission approved recommendations from the near-term task force report, what language would you recommend?

Response: As a threshold matter, SNC believes that a license condition is not necessary to ensure effective and timely implementation of any future Commission-approved recommendations issued in response to the near-term task force report. As SNC's testimony in the mandatory hearing indicates, NRC regulations fully and adequately provide the Commission with a number of options for imposing new requirements arising from the recommendations of the near term task force report subsequent to issuance of the Vogtle 3 and 4 COLs. The options include those applicable to the operating fleet, such as amendments to Commission regulations and the imposition of plant-specific orders amending a license. Those options not only benefit from the Commission's deliberative processes and procedural requirements, but also authorize the Commission to take any actions relative to either the AP1000 DCD or the Vogtle COLs as long as the Commission finds that the new requirements are necessary to adequately protect public health and safety or otherwise satisfy Commission regulations. *See Pacific Gas And Elec. Co.* (Diablo Canyon Power Plant Indep. Spent Fuel Storage Installation), CLI-02-23, 56 NRC 230, 240 (2002); *Duke Cogema Stone & Webster* (Savannah River Mixed Oxide Fuel Fabrication Facility), CLI-01-28, 54 NRC 393, 400 (2001); *Private Fuel Storage, L.L.C.* (Indep. Spent Fuel Storage Installation), CLI-01-26, 54 NRC 376, 383-84 (2001); *see generally* 42 U.S.C. § 2201(b) (2006); 10 C.F.R. §§ 2.202, 52.63(a)(1), 50.109, 52.98.

The options for amending a license in the NRC's regulations are preferable from a regulatory perspective to a license condition that requires the licensee to comply with as yet unspecified requirements because they not only afford greater certainty to the licensee and the NRC as to what the new requirements are, but they also are subject to a more well-defined process in the NRC's regulations for identifying the requirements in question. SNC believes that such a license condition should include language that provides that any requirement to be imposed on the licensee be described in a regulation or an order issued pursuant to Commission regulations.

Notwithstanding the foregoing, in the event the Commission concludes that it is necessary to impose a license condition requiring implementation of Commission-approved recommendations from the near-term task force report, the condition should specify the process for how the requirements applicable to the licensed facility will be identified, including how the Commission's procedural requirements applicable to backfits and amendments to licenses will be observed.

No.	Category	Reference	Directed To	Question
2	Safety	General	Applicant	What process is the industry using on an ongoing basis to factor operating experience from across the world into construction best practices?

Response: In general, the industry is cooperating with the Institute of Nuclear Power Operations (INPO) for the sharing of construction experience (CE). Specifically for Vogtle Units 3 and 4, in addition to working with INPO, there are other mechanisms in which CE is obtained. SNC participates in a program with AP1000 Owners Group (APOG), a group of utility owners with common plans to construct and operate nuclear facilities utilizing the AP1000 design, to monitor and evaluate CE in a consistent fashion. SNC has a site-specific program in place for monitoring and evaluating CE (which accounts for NRC generic communications), including the Corrective Action Program and the establishment of resources in China and agreements with the Chinese utilities to allow direct access to construction progress of the AP1000 units currently under construction in China.

Additionally, the Vogtle constructor and architect engineer, Westinghouse (WEC) and Stone & Webster (together, "Consortium"), collect and evaluate CE resulting from other projects they support, such as the AP1000 plants being built in China. There are a number of processes that WEC and Stone & Webster are both using to factor operating experience into construction best practices, including the Corrective Action Program, a lessons learned program, and the design change process.

These processes are expected to continue throughout the construction phase of the Vogtle Units 3 and 4 project.

No.	Category	Reference	Directed To	Question
7	Safety	Emergency Plan	Applicant	What is the relationship between the Savannah River Site and the Vogtle plants with respect to radiological protection? How does the Vogtle emergency plan address nuclear workers at the Savannah River Site? Are they considered nuclear workers or members of the public? Are they evacuated with the general public?

Response: For clarity, SNC has provided a response to each discrete question contained in Question 7.

What is the relationship between the Savannah River Site and the Vogtle plants with respect to radiological protection?

The relationship between the Savannah River Site (SRS) and the VEGP is described in the VEGP emergency plan. In the event that an emergency is declared at VEGP, SRS would be promptly notified (within 15 minutes), and SRS would take actions in accordance with their emergency plan and emergency implementing procedures. The actions that SRS could take in the event of a declared emergency at VEGP are outlined in a memorandum of agreement between the DOE, SRS and SNC.

SRS would respond to a declared emergency at VEGP as follows:

a. Provide for the prompt notification of all persons on SRS within VEGP's plume exposure pathway Emergency Planning Zone;
b. Assess radiological hazards on SRS and decide upon and implement any protective actions necessary to protect the health and safety of affected persons on SRS, including access control;
c. Perform radiological monitoring as requested by SNC or the State of South Carolina and provide monitoring results to SNC and to the States of South Carolina and Georgia;
d. Provide resources and support as indentified in the Federal Radiological Response Plan to address ingestion pathway concerns;
e. Provide meteorological data to SNC, as requested;
f. Advise SNC and the States of South Carolina and Georgia of public information activities concerning the SRS to the maximum extent possible, and provide a spokesperson to the VEGP Emergency News Center when significant media/public interest in SRS activities is anticipated;
g. As the Regional Coordinating Office for DOE Region 3, respond to requests for radiological assistance from SNC, the NRC, or the States of South Carolina or Georgia in the event of an incident involving the actual or potential release of radiological materials. This assistance will be provided under the Radiological Assistance Program (RAP) and will be limited to technical advice and resources for monitoring and assessment actions essential for the control of the immediate hazards to health and safety. DOE radiological assistance will be terminated when it is no longer needed or the necessary assistance is available from State, local, or commercial services; and

h. As the Regional Coordinating Office for DOE Region 3, advise SNC, the NRC, or the States of South Carolina or Georgia of additional DOE Emergency Response assets available to assist in the response.

How does the Vogtle emergency plan address nuclear workers at the Savannah River Site?

Consideration of the radiological protection of nuclear workers at SRS is integrated throughout the VEGP emergency plan.

The preamble to the emergency plan notes that the major portion of the plume exposure pathway emergency planning zone (EPZ) in South Carolina is within the DOE SRS. The Department of Energy, Savannah River Operations Office (DOE-SR), pursuant to a memorandum of agreement between Georgia Power Company (GPC), as assigned to SNC, and DOE-SR, will be responsible for all emergency response actions on the SRS whenever an emergency occurs at the VEGP.

Section A.10.4 of the emergency plan, *Savannah River Site*, which assigns responsibilities to organizations affected by the emergency plan, assigns responsibility to DOE-SR to provide the necessary response within the SRS reservation in accordance with the SRS emergency plan. In addition, DOE will exercise overall responsibility, jurisdiction, and authority for conducting on-plant response operations to protect the health and safety of SRS personnel.

Also, DOE will provide for emergency notification and, as needed, evacuation, monitoring, decontamination, and immediate life saving medical treatment of non-SRS personnel on the plant site. DOE will also provide access control for SRS areas.

DOE will also provide initial radiological monitoring and assessment support to the State of South Carolina under the DOE RAP. This includes projected release dispersion information and offsite radiological monitoring and assessment assistance. SRS will also coordinate public affairs activities with the State of South Carolina, SNC and GPC. By memorandum of agreement between DOE-SR and GPC, as assigned to SNC, DOE will provide radiological monitoring within about 10 miles of the VEGP site in the State of South Carolina.

Section E.4 of the emergency plan, *Notification of the Public*, notes that DOE-SR has agreed to provide for the prompt notification of all persons on the SRS within the Site's plume exposure pathway EPZ.

Section F.2.1 of the emergency plan, *State of South Carolina,* which describes the methods of communication for affected organizations within the State of South Carolina, notes that the primary means of communication between the Vogtle site and South Carolina is the Emergency Notification Network (ENN), a dedicated telephone system from the Vogtle site to South Carolina emergency response agencies. The ENN has multiple drops in South Carolina state facilities which will be located at the South Carolina Warning Point (the State Emergency Operations Center (SEOC)) which is manned on a twenty-four seven basis. Commercial telephones provide the backup for the ENN. An Administrative Decision Line connects the Emergency Operations Facility (EOF), the SRS Operations Center, the Georgia Emergency Management Agency forward emergency operations center, the SEOCs of both states, and the three South Carolina counties. The section also notes that the State Emergency Preparedness

Director will be responsible for communication at the SEOC with the Vogtle site, SRS, and contiguous local and State governments.

Section I.5 of the emergency plan, *Field Monitoring,* notes that, depending on wind direction and/or the severity of the incident, additional field monitoring teams may be provided by the Department of Natural Resources, South Carolina - Department of Health and Environmental Control, DOE-SR or other divisions of the DOE. Coordination of these teams and data transfer will be accomplished using existing communication links. The State and VEGP field monitoring teams will be coordinated from the EOF by the Dose Assessment Manager to assure a fully coordinated effort. DOE-SR will direct the field monitoring teams of the SRS depending upon the wind direction. DOE-SR will make their monitoring data available to VEGP and State and local representatives at the EOF. The Dose Assessment Team at the EOF will collate field monitoring data for VEGP dose projection purposes. This information will be available to the State and local representatives at the EOF and to DOE-SR.

Section P of the emergency plan, *Responsibility for the Planning Effort*, notes that the Emergency Planning Supervisor performs a review of the emergency plans for SNC once each calendar year. The review includes a comparison for consistency of all emergency plans for the specific sites including the Security Plan, State, County, and the SRS plan as appropriate.

Are [workers at the Savannah River Site] considered nuclear workers or members of the public?

DOE-SR personnel would be considered nuclear workers and would be protected from radiological hazards in accordance with the SRS emergency plan. Also, DOE-SR will provide for emergency notification and, as needed, evacuation, monitoring, decontamination, and immediate life saving medical treatment of non-SRS personnel.

Are [workers at the Savannah River Site] evacuated with the general public?

DOE-SR personnel would be evacuated if so directed under the provisions of the DOE-SR emergency plan. Members of the general public within the VEGP plume exposure EPZ outside of the area under the control of the DOE-SRS site would be evacuated in accordance with the direction required by the respective State and local emergency plans.

No.	Category	Reference	Directed To	Question
8	Safety	Physical Security Plan	Applicant	We understand that there are no NRC regulatory requirements for the physical security plan during the construction phase and fabrication of components. However, what measures are being taken to assure security at the site during construction? What is being done for receipt inspection of components that are received on site or the fabrication of components off site? How will you implement the transition from construction to operation? What changes will occur in the security to initially establish a secure site?

Response: Vogtle Units 3 and 4 are implementing a Site Specific Industrial Security Plan to put into place security measures during construction. The plan follows the guidance in Nuclear Energy Institute (NEI) 09-01 and the rule requirements in 10 C.F.R. Part 26. NEI 09-01, "Security Measures During New Reactor Construction," dated September 2009, was developed by the NEI New Plant Security Task Force to assist licensees in developing plans and procedures for physical security of the construction site during the construction of a new reactor plant. It outlines principles of physical security measures, such as physical barriers, postings, security force monitoring and surveillance, communications and interfaces with local law enforcement, and when applicable, with security forces of operating plant(s) in the same owner controlled area. It also discusses elements of construction site access, such as personnel screening, badging programs, control of visitors, vehicular access, access for emergency responders, control of explosives, and reporting of suspicious behaviors. Site access approval and fitness for duty screening for the construction workforce and qualifications for security personnel are also discussed.

The Security Measures in NEI 09-01 and the site security plan being implemented by Vogtle Units 3 and 4 describe actions to be taken during nuclear power plant construction to detect and deter conditions that would impair the capabilities of security- and safety-related structures, systems, and components (SSCs) to perform their intended functions, where those conditions would not otherwise be detected by the numerous other regulatorily required barriers to ensure that SSCs are installed and tested as designed. Some of these barriers include: receipt inspections, ITAAC implementation, Quality Control/Quality Assurance reviews and inspections, system testing and turnover, extensive pre-installation preparations and in-situ non-destructive testing, foreign material exclusion controls, and monitoring of workers and their products by supervision.

For personnel access, screening is performed with a graded approach based on employer, 10 C.F.R. Part 26 requirements, and job function. For the primary constructor personnel, there are three definitive categories that determine the nature and extent of the pre-access screening.

(1) Individuals functioning as Fitness for Duty (FFD) Program administrators

These individuals are required to submit Self-Disclosure for Suitable Inquiry per Part 26 and receive a background check by a third-party vendor that includes:

- verification of true identity;
- a criminal history check that includes:
  - national database search
  - warrant check for counties of residence listed on application for 7 years;
- an employment history verification;
- an education history verification;
- credit check; and
- vetting through an employment company known persons list or bar list.

In addition, they are subject to a Minnesota Multiphasic Personality Inventory (MMPI) and once the demographic check is in place through the NRC they will be submitted for vetting through the Terrorist Screening Center (TSC).

(2) Management & Supervision subject to Part 26 Subparts A-H, N, and O

These individuals are required to submit Self-Disclosure for Suitable Inquiry per Part 26 and receive a background check by a third-party vendor that includes:

- verification of true identity;
- a criminal history check that includes:
  - national database search
  - warrant check for counties of residence listed on application for 7 years;
- an employment history verification;
- an education history verification;
- vetting through an employment company known persons list or bar list; and
- once the demographic check is in place through the NRC, they will be submitted for vetting through the TSC.

(3) Craft & Administrative personnel subject to Part 26 Subpart K

Salaried individuals are required to submit Self-Disclosure for Suitable Inquiry per Part 26 and receive a background check by a third-party vendor that includes:

- verification of true identity;
- a criminal history check that includes:
  - national database search
  - warrant check for counties of residence listed on application for 7 years;
- an employment history verification;

- an education history verification;
- vetting through an employment company known persons list or bar list; and
- once the demographic check is in place through the NRC, they will be submitted for vetting through the TSC.

Non-salaried individuals (skilled craft labor) receive a background check by a third-party vendor that includes:

- verification of true identity;
- a criminal history check that includes:
    - national database search
    - warrant check for counties of residence listed on application for 7 years
    - vetting through an employment company known persons list or bar list; and
- once the demographic check is in place through the NRC, they will be submitted for vetting through the TSC.

Security force personnel must meet Pre-Access Screening, FFD, and Security Force Qualifications as listed in NEI 09-01, which address age, citizenship, education, criminal history, registrations, licensing, and certifications in accordance with state law.

Vehicle and material access is controlled using a check point to ensure deliveries are authorized. Each delivery is checked to be sure the material being delivered has the proper paperwork and the vehicle is routed to the appropriate receipt location. At the receipt location, the materials are inspected per the purchase documents to assure the material is authorized, has not been damaged and meets the requirements of the purchase agreement. Receipt inspection activities are conducted based on risk to nuclear safety of the components and materials. Quality Control conducts oversight of material receipt, storage and handling using a risk-based approach considering the component's nuclear safety significance. Any anomalies are noted and placed into the corrective action program. Components fabricated offsite will also go through a receipt inspection. The large items will need a pre-installation preparation performed onsite and this again will be an opportunity to discover any anomalies.

Construction security will continue until an operational Protected Area is declared. Prior to the receipt of new fuel assemblies onsite, a portion of the facility will be declared as a Controlled Access Area per the Special Nuclear Material Physical Protection Program that was submitted as part of the COL application. This area will be locked down for the receipt and storage of the new fuel assemblies and will be protected by security personnel that are employed by the Applicant and are trained in accordance with the Applicant Physical Security Plan (PSP). Once the facility reaches the point of systems completion such that an operational Protected Area can be declared, then the PSP that was submitted with the COL application will be implemented. This plan meets the requirements of 10 C.F.R. § 73.55 and facility personnel and material access controls and physical security measures will switch over to the PSP. At this point, the construction security measures will be upgraded to 10 C.F.R. § 73.55 security standards for the unit that is moving toward loading fuel and the Applicant will be implementing security on that unit to prepare for loading fuel. The facility will go through a security sweep and the unit will be locked down for final testing prior to loading fuel.

The unit that is still under construction will remain under the construction security program until the point where systems completion will allow the declaration of an operational Protected Area for that unit. The second unit will go through the same transitional process as the first unit as it moves toward completion and fuel load.

No.	Category	Reference	Directed To	Question
9	Safety	SER Chapter 3	Staff or Applicant	In April 2011, the DNFSB sent a letter to DOE citing concerns with a computer code used for building analysis. This letter is publically available on the DNFSB web site and explains that the computer code is used both by DOE for defense nuclear facilities and by the commercial nuclear power industry. The computer code is called SASSI and is used for evaluation of SSI effects between the building and its supporting soil. The letter states (at p. 1): Recently, SASSI users have identified significant technical and software quality assurance issues with this software. In August 2010, the Los Alamos National Laboratory . . . published LA-UR-10-05302, Seismic Response of Embedded Facilities Using the SASSI Subtraction Method, identifying issues with the SASSI subtraction method, which is extensively used in DOE's design and construction projects. The [DNFSB] is concerned that these issues could lead to erroneous conclusions that affect safety-related structural and equipment design at DOE defense nuclear facilities. Did building designers for this application use this computer code? Do the concerns cited by the DNFSB affect the Vogtle building design?

Response: WEC is the building designer of record. The seismic loads used in the AP1000 building design are based on the AP1000 design certification 3D generic soil structure interaction analyses. The AP1000 3D generic soil structure interaction analyses are not affected by the concerns with the SASSI subtraction method.

For hard rock sites, WEC used a computer program called ANSYS for the AP1000 design. For the assessment required to address Interim Staff Guidance – 1 that hard rock, high frequency motions did not govern the design, WEC has used a NI20 SASSI hard rock surface model (not embedded). Thus, the concern with SASSI is not applicable for hard rock sites.

The SASSI analyses for all the generic soil conditions, including soft soil, were performed using the SASSI direct method; not the subtraction method. During a structural review meeting with the NRC, the subtraction method issue was discussed, and it was confirmed that the AP1000 analyses are not impacted by the issue.

Thus, the concern with the SASSI subtraction method is not applicable for any of the AP1000 soil structure interaction analyses used by WEC in support of the AP1000 Design Certification amendment.

The AP1000 Nuclear Island (NI) is a standard design; and therefore, not designed for a specific site. The COL applicant has to demonstrate that the standard design is acceptable for the COLA site conditions.

To demonstrate this, 3D SASSI analyses were performed. The applicant choose to refine the standard AP1000 SASSI model, called the NI20 model, to a finer finite element (FE) mesh size for the embedded portion of the NI. This site-specific SASSI model is called the Vogtle NI15 model. This was done due to the GMRS exceedances of the Certified Seismic Design Response Spectra (CSDRS) and the fact that the Vogtle site is a deep soft soil site that is significantly different than the generic soil profiles used to develop the AP1000 CSDRS design in-structure response spectra (ISRS) envelope. The NI15 structural model was benchmarked against the standard AP1000 NI structural models to assure structural and dynamic equivalence. Due to the finer FE mesh size and the softer and deeper soil profile, *a modified* subtraction method was used to allow the much larger SASSI model to run efficiently. The Vogtle 3D SASSI analyses did not use the subtraction method.

The concerns cited by the DNFSB were known and considered. In regards to these concerns, the AP1000 NI design at the Vogtle site is acceptable based on the following:

1. A letter dated July 29, 2011 from DOE to DNFSB (Reference 1) provided a response to the DNFSB letter that citing concerns with the SASSI subtraction method. The report attached to the DOE July 29, 2011 letter provided documentation that if the solution using the subtraction method is not reasonable, an alternate method called the “modified subtraction method” may be used. The modified subtraction method is shown to be in good agreement with the computationally much more intensive direct method results over a wide frequency range. Therefore the modified subtraction method was used in the SASSI NI15 SSI analysis for the Vogtle site.
2. The structural demand in a building is typically dominated by the response of the building at its fundamental SSI frequency. The Vogtle horizontal fundamental SSI frequency is around 2Hz which is well below the first mode frequency of the excavated soil volume where, at that frequency and above, there is concern with using the subtraction method. Therefore, even though the subtraction method was not

used the subtraction method would not have affected the overall seismic demand on the NI building structure.

3. The AP1000 seismic analyses for the standard design consider a range of generic site conditions including hard rock. The hard rock condition in general produced higher ISRS in the higher frequency ranges. In these frequency ranges the comparison of the VEGP 3D SASSI ISRS to the AP1000 CSDRS design ISRS envelope demonstrated that the AP1000 CSDRS design ISRS envelope is significantly higher; therefore, significant seismic design margin is provided for the design of systems and components mounted in the NI as compared to that required for the Vogtle site-specific seismic demand. Therefore, this large margin provides additional assurance that the AP1000 NI standard design is acceptable for the VEGP site.

Reference 1: Letter from the Deputy Secretary of Energy, Daniel B. Poneman to the Honorable Peter S. Winokur Chairman Defense Nuclear Facilities Safety Board dated July 29, 2011 with attachment: "U.S. Department of Energy Soil-Structure Interaction Report July 2011," Dr. Brent Gutierrez, PE; U.S Department of Energy, Savannah River Operations Office.

No.	Category	Reference	Directed To	Question
12	Safety	SER Chapter 6	Staff	The ACRS recommended a technical specification to ensure containment cleanliness does not compromise sump operability for long term cooling. The Staff's response to the ACRS was to change the cleanliness requirement from Tier 2 to Tier 2*, which will require NRC approval to change. In so doing, the Staff allowed a sampling to be performed on the containment for cleanliness after an outage and the results will be evaluated post-start up. The corrective action program will be used to address any deficiencies. This is described in FSER section 6.3.4 in response to STD COL 6.3-1. Why did the Staff not implement the ACRS recommendation such that containment cleanliness would be assured prior to start-up?

Response: While this question is directed to the Staff, the Applicant offers some additional information. This item was addressed in part in the closing testimony of Joseph A. Miller and is recorded at pages 347-348 of the Transcript. Additionally, the FSAR states:

> The sampling is conducted after containment exit cleanliness inspections to provide reasonable assurance that the plant latent debris design bases are met. Sampling frequency and scope may be adjusted based on sampling results. Results are evaluated post-start up and any nonconforming results will be addressed in the Corrective Action Program.

As a point of clarification, the sampling occurs after containment exit cleanliness inspections *during* the outage, not *after* the outage as indicated in the question.

The intent of the FSAR description was to emphasize that if there is a visual confirmation that the debris limits are met, the licensee would not be constrained from starting back up to wait on confirmatory laboratory analysis results of the samples. The FSAR also indicates that the guidance of NEI 04-07 will be used to conduct the sampling program. That guidance indicates that "the [analysis] work should be performed by a laboratory experienced in material identification, analysis of the macroscopic and microscopic properties of material samples, and handling of radioactive materials." The resources to perform this type of analysis may or may not exist onsite and therefore, the sample could possibly have to be sent to an offsite laboratory. If there is clear visual evidence that the debris limits are being met, there is no need to constrain the licensee from moving forward in proceeding to full power. If, on the other hand, visual evidence reveals any question regarding compliance with the debris limits, the plant would remain shut down until such time that the design basis to support operation could be restored. Plant procedures will implement these requirements to assure the safe operation of the plant.

No.	Category	Reference	Directed To	Question
13	Safety	SER Chapter 15	Staff or Applicant	In Chapter 15 of the Staff's FSER there is a discussion of the LEFM. There are also two license conditions related to determination of power calorimetric uncertainty and there is an ITAAC to assure the overall instrumentation uncertainty is less than the safety analysis uncertainty of 1%. There is little discussion of this in the application, but FSER p. 15-4 describes commitments that if the LEFM fails the plant will de-rate and use the feedwater flow venturi to ensure power is within safety analysis and uncertainty limits. How will the Applicant reconcile differences between the feedwater flow venturi and the LEFM if they are not consistent? Does the Staff expect the Applicant to monitor the differences between the two feedwater flow instruments through power ascension testing? There is a commitment in the FSER on p. 15-4 to perform periodic calibration on instrumentation used as inputs to the calorimetric. There is also the commitment to de-rate if the LEFM fails and to use the feedwater venturi instead. Since the Staff's safety evaluation is not used to require compliance—why are these not captured as commitments in the application?

Response: The Applicant provides the following statement in response to the first question identified above, *i.e.*, "How will the Applicant reconcile differences between the feedwater flow venturi and the LEFM if they are not consistent?" The remaining questions are understood to be directed to the NRC Staff.

The feedwater system ultrasonic flow meters (UFM) are specified to provide a maximum uncertainty of ± 0.5% of mass flow rate. The mass flow rate calculated by the UFM is used to normalize the high range venturis. The normalized venturis are then considered the primary feedwater flow measurement for use in the calorimetric calculation. Absent any fouling or erosion of the venturi piping, the UFM and venturi flow meters are expected to correlate nearly one to one at high power, steady state operation. Therefore, the initial value of the normalization factor is expected to be approximately 1.00 (*i.e.* $N_F = W_{UFM} / W_{Venturi} = 1$), where N_F is the normalization factor, W_{UFM} is the UFM mass flow rate, and $W_{Venturi}$ is the venturi mass flow rate).

The normalization factor is calculated for each UFM-venturi set at 1 minute intervals, at greater than 70% rated thermal power. When the normalization factor differs from its programmed value by a predetermined amount, the calorimetric program provides an input to the Plant Control System for generating an alarm in the main control room and at the remote shutdown workstation. The alarm indicates the normalization factor is outside expected limits, and requires operator action to address the issue.

Potential sources creating differences between the UFM mass flow rate, and the venturi mass flow rate include, but are not limited to: 1) a faulted Main Feedwater System temperature and/or pressure sensor; 2) a faulted UFM transducer; and 3) a change in pipe geometry (*e.g.* venturi fouling, pipe dimensions, alignment, thermal expansion, etc.). A change in any of the parameters above would affect the UFM mass flow rate, the venturi mass flow rate, or potentially both. The operator is expected to review all appropriate parameters and determine the cause of the difference between the two flow rates. Once a determination has been made, the appropriate actions can be implemented to reconcile the difference between the measurements (*e.g.*, remove the UFM from service for maintenance, remove a sensor from service for maintenance, recalculate the normalization factor, etc.).

No.	Category	Reference	Directed To	Question
14	Environmental	Tr. at 325-26	Applicant	Please describe your analysis of the environmental impacts of the Fukushima events. Identify the relevant information you drew from the task force report and any other sources and describe your analysis of that information and your conclusions.

Response: SNC uses the New and Significant Process as described in Response to Question 35(c) provided in SNC's September 13, 2011 filing to the Commission's Order of August 31, 2011 for evaluating whether information available since the Vogtle ESP Environmental Impact Statement (EIS) is New and Significant.

The Fukushima event that occurred March 11, 2011 was a tragic and unexpected event that has been the topic of much discussion and investigation. As information became available, SNC considered whether the information warranted inclusion in the New and Significant Process. Since the event, SNC reviewed available information in the form of media articles, information released by the NRC, such as memorandums and the Task Force Report. Based on the totality of our review, SNC determined the Fukushima event consisted primarily of two incidents: first, the natural phenomena or initiating event (earthquake and tsunami) and second, the severe accident.

After screening available information, SNC determined that no documented "New and Significant" evaluation of the Fukushima event was necessary because the event did not screen in to the process as "new" information under 10 C.F.R. § 51.50(c)(1) as defined by SECY2006-0220. According to SECY2006-0220, for information to be new it must be both:

> (1) not considered in preparing the ESP environmental report or EIS (as may be evidenced by references in these documents, applicant responses to NRC requests for additional information, comment letters, etc.) and
>
> (2) not generally known or publicly available during the preparation of the EIS (such as information in reports, studies, and treatises).

The following paragraphs explain the logic.

As previously indicated, the Fukushima event consisted of the natural phenomena that were the initiating events (earthquake and tsunami) and the severe accident. Both were considered during the VEGP ESP review and included in the COL. Specifically, natural phenomena with the potential to impact the site (*e.g.*, probable maximum flood, dam breaks, seismology, etc.) are described in the ESP Safety Evaluation Report in Sections 2.4 and 2.5. The environmental consequences of severe accidents are evaluated in the ESP EIS in Section 5.10 for single unit accidents and Section 7.9 for cumulative impacts of severe accidents. The COL Supplemental EIS concluded there was no available New and Significant information at the time it was published and the conclusions in the ESP EIS remained valid.

For the EIS severe accident assessment, the consequences of an accident are considered independently of the initiating event. This analytical approach is consistent with the guidance in NUREG 1555, Environmental Standard Review Plan, which was used to develop the Vogtle Units 3 and 4 environmental impacts of postulated accidents involving radioactive materials.

No information available from the NRC Task Force Report indicates that the earthquake or tsunami in Japan could invalidate the assumptions utilized in either the Vogtle FSAR or the ESP Final EIS and COL Final Supplemental EIS.

EISs are detailed documents required by the National Environmental Policy Act (NEPA) (Section 102(2)(c)) and prepared for all major Federal Actions that significantly impact the human environment (40 C.F.R. § 1508.11). Specific to the requirements of NEPA, an EIS must identify all indirect effects that are known, and make a good faith effort to explain the effects that are not known but are "reasonably foreseeable" (40 C.F.R. § 1508.8(b)).

SNC believes that initiating events such as those in Japan are not "reasonably foreseeable" at the VEGP location, based on the analysis of VEGP specific site characteristics done for the ESP and COL, and that severe accidents for the selected reactor technology (AP1000) have been evaluated in the VEGP EIS. Similarly, the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident states that, "a sequence of events like those occurring in the Fukushima accident is unlikely to occur in the United States…" and, that "the AP1000 design has many of the design features and attributes necessary to address the Task Force recommendations."

Additionally, SNC is providing related clarification to the discussion responding to a question regarding the impact analysis of an accident involving multiple units at a single site during the environmental question and answer portion of the Mandatory Hearing Environmental Panel 1 on September 28, 2011. The question and answer are recorded immediately after the exchange referenced in Item M at p. 328 lines 23-25 and p. 329 lines 1-12. Because this question and answer overlaps with discussion related to the New and Significant consideration SNC undertook with respect to Fukushima (*see* p. 329 lines 10-12), SNC includes this clarification with this response to Question 14.

The overall risk to the public from the four unit site was discussed in the ESP EIS (Section 7.9). The risks described in Section 7.9 of the ESP EIS do include the potential, however extremely remote, that two or more reactors could experience concurrent accidents caused by different, independent events. However, the cumulative dose to the population from a severe accident involving multiple units at a single site was not evaluated in the ESP Environmental Report developed by SNC.

The cumulative impacts of a severe accident presented in Section 7.9 of the ESP EIS are derived by combining the four estimated reactor population dose risks for each reactor at or planned at VEGP. The cumulative impact is presented as the summation of the Population Dose Risks associated with a severe accident at each of the units. However, regarding the risk from multiple unit accidents, the EIS discussion assumes the risk from the reactors are independent, meaning that there is no credible event or series of events that could increase the likelihood of multiple concurrent accidents. This simplified analysis recognizes the dominant effect of adding units to an existing site (*e.g.*, that the risk to the population is essentially a multiple of the number of reactors at the site). The population dose risk of an accident from a single reactor was discussed in Section 5.10.2, *Severe Accidents*, of the ESP EIS.

The severe accident impact analysis results in a measure of the Population Dose Risk, which considers the radiation exposure, or dose (consequences) to the population, and the frequency of the accident.

The risk from concurrent accidents would be a function of two factors, the increased consequences to the public caused by larger potential inventory of radionuclides, combined with the lower frequency of multiple unit accidents. It is reasonable to expect that for a site having Vogtle's limited **external hazard** exposure the risk to the public would remain dominated by the risk from an accident at any single unit. Therefore, the cumulative severe accident impact to population dose risk associated with four reactors at Vogtle remains SMALL.

Respectfully submitted,

(Original signed by M. Stanford Blanton)

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Dated this 17th day of October, 2011.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	)	
Southern Nuclear Operating Company	)	**Docket Nos. 52-025-COL and 52-026-COL**
(Vogtle Electric Generating Plant, Units 3 and 4)	)	**October 17, 2011**

CERTIFICATE OF SERVICE

I hereby certify that copies of EXHIBIT SNC000011 for the Vogtle Units 3 & 4 COL Mandatory Hearing in the above-captioned proceeding have been served by electronic mail as shown below, this 17th day of October, 2011, and/or by e-submittal.

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