UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	)	
	)	
SOUTHERN NUCLEAR OPERATING CO.	)	Docket Nos. 52-025-COL and 52-026-COL
	)	
(Vogtle Electric Generating Plant	)	
Units 3 and 4)	)	
	)	

NRC STAFF RESPONSES TO COMMISSION POST-HEARING QUESTIONS

Pursuant to the Commission's Order (Supplemental Responses and Post-Hearing Questions) of October 6, 2011, the staff of the U.S. Nuclear Regulatory Commission ("Staff") hereby responds to the questions posed in that Order. These questions provide supplemental responses to questions posed by the Commissioners during the hearing as well as answers to additional post-hearing questions.

The Commission's Order directed some questions to the Staff, some to Southern Nuclear Operating Company ("Applicant," "SNC"), and some to both. Attachment A to this filing presents the Staff's responses. Where a question was directed to both the Staff and Applicant, the Staff's response is included in the attached; however, where a question or sub-question was directed solely to the Applicant, the Staff accordingly has not provided a response.

/Signed (electronically) by/
Patrick A. Moulding
Counsel for the NRC Staff
U.S. Nuclear Regulatory Commission
Mail Stop O-15 D21
Washington, D.C. 20555-0001
(301) 415-2549
(301) 415-3725 fax
Patrick.Moulding@nrc.gov

Dated at Rockville, Maryland
This 17th day of October, 2011

ATTACHMENT A

NRC Staff Responses to Commission

Post-Hearing Questions

NRC Staff Responses to Commission Post-Hearing Questions

Table 1 – SUPPLEMENTAL RESPONSES TO IN-HEARING QUESTIONS

For clarity with respect to several of the items in Table 1, the staff has excerpted the question or, where necessary, summarized what it understands the follow-up question to be.

Item A

Staff Overview Panel, p. 59, lines 24-25; p. 60, lines 1-4

What is the status of COL action items, and are they incorporated as commitments or license conditions?

Staff Response:

A description of how all COL action items (COL information items) were addressed and, as appropriate, captured in the Vogtle licensing basis, is included in the response to Post-Hearing question No. 3, below.

Item B

Staff Overview Panel, p. 79, lines 13-25; p. 80, line 1

Given the perspective of a public citizen in the area looking at Federal activities, and who may not distinguish, for example, between the NRC or DOE, is there a rough philosophical alignment between the environmental approaches of the agencies that have big footprints in that area?

Staff Response:

The staff responded to a similar question on Day 2 of the mandatory hearing (see Tr. at 342-343). With respect to the development of the environmental impact statement, the NRC's approach is fundamentally consistent with that of other Federal agencies. Other agencies may differ in how they approach certain aspects of the review, such as scoping. Agencies may also place different emphasis on certain resource areas because of their specialized regulatory jurisdiction; for example, the U.S. Army Corps of Engineers has detailed regulatory criteria by which it examines potential impacts to wetlands and waters of the United States as a result of its statutory responsibilities under Section 404 of the Clean Water Act and Section 10 of the Rivers and Harbors Act of 1899. However, there is broad alignment across the agencies with which the NRC interacts in its reviews regarding the underlying approach to implementing NEPA.

Item C

Safety Panel 1, p. 121, lines 18-25

How much in advance of fuel load would the licensee have to establish and fund its Decommissioning Trust?

Staff Response:

In accordance with 10 C.F.R. § 50.75(e)(3), Southern Nuclear Operating Company, after issuance of the combined licenses, will submit a report for each unit, no later than thirty (30) days after the NRC publishes notice of intended operation in the *Federal Register* pursuant to 10 C.F.R. § 52.103(a). This report will contain a certification that financial assurance for decommissioning is provided in the amount specified in SNC's most recent updated certification, including a copy of the financial instrument to be used.

Item D

Safety Panel 1, p. 123, lines 10-25; p. 124, lines 1-4

If an applicant operated a nuclear plant and, hypothetically, it had a troubled history, would that be a factor in the staff's evaluation?

Staff Response:

As the staff explained in the Vogtle FSER, because SNC holds 10 CFR Part 50 licenses for nuclear power plants (SNC operates the Edwin I. Hatch Nuclear Plant, Units 1 and 2, the VEGP Units 1 and 2, and the Joseph M. Farley Nuclear Plant, Units 1 and 2) and has demonstrated its ability to build and operate these plants, the staff finds that SNC is qualified to hold a 10 CFR Part 52 license. This includes SNC's demonstrated ability to choose and manage oversight of NSSS vendors, architect engineers and constructors of nuclear related work. The staff also noted that Section 17.5 of the VEGP COL FSAR discusses the QA program to be implemented at the receipt of the COL. This QA program includes requirements that will be implemented by SNC's NSSS vendor, architect engineer, and constructor. The staff's evaluation of Section 17.5 of the VEGP COL FSAR is in Section 17.5 of this SER. Based on SNC's experience with nuclear power plants and the staff's evaluation of SNC's QA program, the staff finds that SNC is technically qualified to hold a 10 CFR Part 52 license in accordance with 10 CFR 52.97(a)(1)(iv).

In general, the fact that an applicant is also a current power reactor licensee provides the necessary support for the staff's finding of technical qualifications under 10 CFR 52.97(a)(1)(iv). Staff review guidance does not specify circumstances under which the staff would need to probe further into such an applicant's technical qualification. However, if during the COL review the staff is (or becomes) aware of aspects of the applicant's experience as a current licensee that appear to be material to that applicant's qualifications with respect to some aspect of the COL application, the staff, in its professional judgment, might find it appropriate to conduct further review of those facts before making its 52.97(a)(1)(iv) conclusion. This approach is consistent with Commission caselaw concerning the adequacy or "integrity" of an applicant/licensee's management or corporate organization, confirming that issues such as past violations of NRC regulations would indicate a deficiency in an application only if they are

directly germane to the licensing action, rather than being of simply historical interest. *See Dominion Nuclear Connecticut, Inc.* (Millstone Nuclear Power Station, Units 2 and 3), CLI-01-24, 54 NRC 349, 365 (2001); *Georgia Inst. of Tech.* (Georgia Tech Research Reactor), CLI-95-12, 42 NRC 111, 120 (1995); *USEC, Inc.* (American Centrifuge Plant), LBP-05-28, 62 NRC 585, 618-19 (2005).

Item E

Safety Panel 2, p. 160, lines 15-19

Is a Tier 2* commitment comparable to a Technical Specification? For example, when sweeps are conducted 15 years into operation and the licensee exceeds the limits, what happens under Tier 2*?

Staff Response:

See Staff Response to Post-Hearing Question Nos. 4 and 12, below.

Item F

Safety Panel 2, p. 164, lines 11-19

If the staff finds an operational program like squib valve in-service testing to be insufficient such that it would not allow start-up of the plant, what's our legal authority to do that at that point if we don't have an ITAAC?

Staff Response:

See response to Question 5.a in Table 2, below.

Item G

Safety Panel 3, p. 231, lines 8-21

Clarify the statement in section 19.55.6.3 of the FSAR that "for site specific conditions the review level earthquake is 1.67 times VEGP GMRS, where the VEGP site specific review level earthquake is 1.67 VEGP GMRS."

This question was directed solely to the applicant. Accordingly, the staff has not provided a response.

Item H

Safety Panel 3, p. 239, lines 6-23

Regarding the accident analysis, staff slide 16 lists several sections that reference plant specific information as opposed to basically being incorporated by reference. Taking the decreased reactor coolant inventory as an example, why is that particular item not bounded entirely by the AP1000 design certification?

Staff Response:

The staff reviewed the site specifications necessary for Sections 15.6 and 15.7. The only site specific information required is the site specific atmospheric dispersion factors. The site specific atmospheric dispersion factors were used in the analysis to confirm that the site specific analysis remained bounded by the analyses performed for the design certification.

Item I

Safety Panel 4, p. 277, lines 12-25; p. 278, lines 1-25; p. 279, lines 1-8

Does Part 52 require the EALs to be submitted in the application? If so, why didn't the applicant need an exemption?

Staff Response:

See Staff Response to Post-Hearing Question No. 6.a, below.

Item J

Safety Panel 4, p. 285, line 25; p. 286, lines 1-25; p. 287, lines 1-14

Is the nature of the substantive protection designated for the technical support center less than that of the units' other critical digital assets?

Staff Response:

No. The Southern Nuclear Company's application for Vogtle uses the term "level" to refer to locations in the applicant's cyber security architecture where a digital system within the scope of Title 10 of the Code of Federal Regulations, Section 73.54, "Protection of Digital Computer and Communication Systems and Networks" (10 CFR 73.54) is located. "Level" does not refer to the amount of protection afforded to a Critical Digital Asset (CDA). CDAs are protected through security controls. Therefore, if an applicant determines that systems in the Technical Support Center (TSC) are CDAs, the TSC would not be protected any less than any other CDA. The same requirements apply to all CDAs regardless of their locations in the applicant's architecture and, as required by 10 CFR 73.54, all CDAs must be adequately protected from cyber attacks up to and including the Design Basis Threat.

Item K

Safety Panel 4, p. 293, lines 22-25

Can you briefly summarize the other initial plant test relevant to station blackout?

Staff Response:

The requirements of 10 CFR 50.63, "Station Blackout," state that nuclear power plants must have the capability to withstand for a specified duration and recover from a station blackout (SBO) event (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system.)

The AP1000 design minimizes the potential risk contribution of an SBO by not relying on ac power sources for design-basis events. The AP1000 safety-related passive systems automatically establish and maintain safe-shutdown conditions for the plant following design-basis events, including an extended loss of ac power sources. The passive systems can maintain these safe-shutdown conditions after design-basis events for 72 hours, without operator action, following a loss of both onsite and offsite ac power sources. On these bases, the staff concluded (in NUREG-1793) that the AP1000 design meets the requirements of 10 CFR 50.63 for 72 hours. DCD Tier 2, Section 1.9.5.4, "Additional Licensing Issue," provides additional information on long-term actions following an extended SBO beyond 72 hours.

The following AP1000 design features mitigate the consequences of an SBO:

- Full load rejection capability to reduce the probability of loss of onsite power
- Safety-related passive residual heat removal heat exchanger
- Safety-related passive containment cooling
- Bleed and feed (i.e., natural circulation) capability, using the safety-related automatic depressurization system in conjunction with the water available from the core makeup tanks, accumulators, and in-containment refueling water storage tank
- Class 1E batteries sized for 72 hours of operation under SBO conditions
- Reactor Coolant Pumps without shaft seals
- Passive cooling for the rooms containing equipment assumed to operate during SBO conditions (the protection and safety monitoring system cabinet rooms and the Main Control Room) so that this equipment continues to operate for 72 hours

There are several key tests that will be performed as part of the initial test program to demonstrate that these AP1000 design features perform as designed:

AP1000 DCD Section 14.2.10.4.24, "Plant Trip from 100 Percent Power"

This test is conducted to verity the ability of the AP1000 plant automatic control systems to sustain a trip from 100 percent rated thermal power (i.e., full load rejection) and bring the plant to stable conditions following the transient.

AP1000 DCD Section 14.2.9.1.3, "Passive Core Cooling System Testing"

The AP1000 Passive Core Cooling System will be tested in accordance with this section. Portions of testing described in Section 14.2.9.1.3 will only be conducted on the first or on the first-three AP1000 plant(s). For example, natural circulation tests using the steam generators

and the passive residual heat removal heat exchangers will be performed at low-core power during the startup test phase of the initial test program for the first-three AP1000 plants (Section 14.2.9.1.3, Items (k) and (w)).

AP1000 DCD Section 14.2.9.1.4, "Passive Containment Cooling System Testing"

This test is conducted to verify, in part, to demonstrate proper system flow rates by draining the passive containment cooling system water storage tank. This testing demonstrates the proper resistance of the four passive containment cooling water storage tank delivery flow paths. This testing also demonstrates that water is supplied at the specified flow rates and times for 72 hours consistent with the design basis analyses presented in AP1000 DCD Section 6.2.1.

In addition, the AP1000 plant will be tested in accordance with AP1000 DCD Sections 14.2.10.4.26, "Loss of Offsite Power," to demonstrate that the dynamic response of the plant is in accordance with the design for a condition of loss of turbine generator coincident with loss of all sources of offsite power.

AP1000 DCD Section 14.2.9.1.14, "Class 1E DC Power and Uninterruptible Power Supply Testing"

This test is conducted to verify, in part, that the AP1000 Class 1E and Uninterruptible Power Supply system provide the required safety-related electrical power for at least 72 hours following a design basis event, independent of both offsite and onsite ac electrical power supplies.

AP1000 DCD Section 14.2.10.4.26, "Loss of Offsite Power"

This test is conducted to demonstrate that the dynamic response of the plant is in accordance with the design for a condition of loss of turbine generator coincident with loss of all sources of offsite power.

AP1000 DCD Section 14.2.9.1.6, "Main Control Room Emergency Habitability System Testing"

This test is conducted to verify, in part, the ability to maintain the main control room environment and temperatures in the protection and safety monitoring system cabinet and emergency switchgear rooms within specified limits for 72 hours (Reference AP1000 DCD Section 6.4.3.2).

Item L

Environmental Panel, 1 p. 325, line 24 (identifying speaker), line 25 (last word, start of question); p. 326, lines 1-13

This question was directed solely to the applicant. Accordingly, the staff has not provided a response.

Item M

Environmental Panel, 1 p. 327, lines 24-25; p. 328, lines 1-6

In its analysis of severe accident impacts in the EIS, did the staff consider more than one reactor unit at a time experiencing a severe accident at the Vogtle site?

Staff Response:

Consistent with current staff review guidance, the severe accident assessment for the environmental review of the Vogtle application does not explicitly evaluate the risk from severe accidents occurring concurrently at more than one unit on the site. With respect to the Vogtle review, the staff had evaluated the environmental impacts of severe accidents in the staff's Final Environmental Impact Statement (FEIS) for the Vogtle Early Site Permit (ESP) (NUREG-1872). In that evaluation, the staff determined that the probability-weighted consequences of severe accidents at the Vogtle site would be SMALL. As part of its assessment, the staff compared the severe accident risks of the new reactors to those of the current-generation reactors, including at other reactor sites, and also relative to the safety goals articulated by the Commission in its Safety Goal Policy Statement. In the supplemental EIS (SEIS) for the combined license (NUREG-1947, p. 5-18), the staff therefore assessed whether there was new and significant information related to this topic. Because it determined that there was not, the staff reaffirmed its ESP-stage conclusion of SMALL impacts.

The staff's guidance on evaluating the consequences of severe accidents in environmental impact statements is provided in Section 7.2, "Severe Accidents," of NUREG-1555, "Environmental Standard Review Plan: Standard Review Plans for Environmental Reviews for Nuclear Power Plants." Among other things, this guidance calls for staff to discuss events arising from causes external to the plant that are considered possible contributors to the risk associated with the plant. The environmental risks of severe accidents are compared to and contrasted with radiological risks associated with normal and anticipated operational releases. However, the guidance does not call for the staff to consider the consequences arising from external causes of severe accidents involving multiple co-located reactor units and spent fuel storage facilities.

Additionally, as part of the ESP FEIS the staff evaluated the cumulative impacts associated with severe accidents by considering the cumulative risk for both the two existing and two proposed units, and found these risks would still be SMALL. The environmental risk associated with two AP1000 reactors in addition to the two existing reactors is the sum of the risks for the four independent individual reactors. As described in the ESP FEIS, the combined population dose risk for the existing two units plus the two new AP1000 reactors is about 3.8×10^{-2} person-Sv/Ryr. The FEIS found that considering the cumulative risk of four units did not constitute a significant increase in the population dose risk. Similar conclusions were obtained for other risks, such as cost risk, early fatalities, and decontamination areas. Ultimately, the staff determined the cumulative severe accident impacts associated with adding two AP1000 reactors to the site to be small. In the COL SEIS, the staff determined that there was no new and significant information that would change its conclusions from the ESP FEIS.

Item N

Environmental Panel, 1 p. 335, lines 10-25; p. 336, lines 1-5

For the severe accidents analyzed in the EIS evaluation and the assumed radiological releases, is it correct that the staff does not expect to see radiological impacts similar to those seen following the accident at Fukushima?

Staff Response:

The staff's examination of severe accidents does consider accidents that, like Fukushima, are assumed to involve radiological releases to the environment. However, to comport with the Commission's policy and NEPA's required focus on reasonably anticipated environmental impacts rather than "worst-case" scenarios, the staff's environmental evaluation of severe accidents considers impacts by evaluating probability-weighted consequences. Because of the potentially high consequence but extremely low probability of such accidents, looking at the consequences without accounting for risk would distort the purpose of disclosing the reasonably anticipated impacts of the project. Because of this framework, while it is clear that severe accidents such as that experienced at Fukushima are potentially high-consequence events, the staff's conclusion in the EIS examines those consequences in terms of risk.

The staff assesses the environmental impacts from severe accidents in terms of its health effects, economic costs, and land affected by contamination (e.g., rem/Ryr, $/Ryr, ac/Ryr). Just as important, the Commission's 1985 Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants directs the staff to describe the impacts from severe accidents in the context of risk. In the case of the Fukushima event, the staff has not completed a PRA or other quantitative analysis of such a multi-unit event occurring as part of the AP1000 design certification probabilistic risk assessment and the safety review of accidents documented in Chapter 19 of the Final Safety Evaluation Report for the Vogtle site. However, as explained above in response to Item M, the staff's analysis documented in the Vogtle COL SEIS (tiering off of the ESP FEIS) considered a range of severe accident scenarios and the associated releases and consequences. Moreover, the Fukushima Daiichi units are BWR-3 and BWR-4 plants with Mark I containments, and as explained in the ESP FEIS, risks calculated for the Westinghouse AP1000 reactor design at the Vogtle site are expected to be lower than those for current-generation plants, supporting the staff's conclusion that the severe accident risks at Vogtle remain SMALL.

Item O

Environmental Panel, 1 p. 336, lines 24-25; p. 337, lines 1-13

In its environmental analysis of severe accidents initiated by an external event, is the staff's impact conclusion based only on the radiological consequences of the accident, rather than the impacts of both the accident and any other damage in the vicinity just from the external event itself?

Staff Response:

The EIS impact conclusion regarding severe accidents does not consider the impact of any other damage in the vicinity caused by the external initiating event (e.g., an earthquake),

because those impacts would not be effects of the Federal action being considered, issuance of COLs for nuclear power plants. As described above in Item M, the severe accidents analysis in the combined license supplemental environmental impact statement (SEIS) relies on, and tiers from, the analysis completed for the early site permit environmental impact statement (EIS). The accidents considered in the EIS are based upon the relevant information provided at the ESP stage for severe accidents and is consistent with accidents evaluated as part of the safety review. Thus, the severe accidents considered in the staff's environmental analysis are based upon the AP1000 design certification probabilistic risk assessment and the safety review of accidents documented in Chapter 19 of the Final Safety Evaluation Report. Specifically, the staff's impact conclusion related to severe accidents in its NEPA analysis is based on the environmental risk due to the radiological consequences of the accident times the probability of the accident to occur. Moreover, based upon the 1985 Commission Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, the staff's EIS describes the impacts from a severe accident in the context of risk. These risks focus on the probability and consequences of the postulated accident, not on independent damage attributable to the external event that may have initiated that accident.

Item P

Concluding Statements, p. 352, lines 10-15

Regarding several questions raised during the course of the mandatory hearing, the staff will provide supplemental responses in accordance with conditional procedures in the direction that is provided by the Commission.

Staff Response:

This document provides the responses to questions raised during the course of the mandatory hearing. No additional response to this item is required.

Item Q

Concluding Statements, p. 362, lines 14-21

We had a long discussion about squib valves yesterday. We didn't ask for anything further on the record but I think I'd like now to formally ask that we get a clearer explanation of what exactly is the situation there, because it seems to me that it's very difficult for us to make the finding that we're being asked to make if we don't know that the situation is stable right now. Not what happens in the future, but can we make a finding based on what we have today. I'd like to get the staff's view on that.

Staff Response:

See response to Question 5.b in Table 2, below.

Miscellaneous Corrections to Staff Testimony

The staff has also identified the following brief corrections to statements made by the staff at the hearing, either in its exhibits or in response to Commission questions.

- On slide 14 of Exhibit NRC000009, the application date listed for the AP1000 DC amendment should be May 26, 2007, rather than March 26, 2007.

- In the staff's response to the Commission's pre-hearing questions 15(a)(i) and (ii), the ADAMS accession numbers (ML081020222 and ML081020207) cited for portions of the Vogtle Early Site Permit application were to Revision 4 of the application rather than Revision 5. The correct accession numbers for those portions of Revision 5 of the ESP application are ML091540887 and ML091540890, respectively.

- A staff witness stated (Transcript at 292, line 6) that Vogtle Unit 1 received its operating license in 1987, and that Vogtle Unit 2 received its operating license in 1987. Unit 2 received its operating license in 1989.

Table 2 – POST-HEARING QUESTIONS

Question 1:

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?

Staff Response:

If the Commission decides that license conditions to implement Fukushima Near-Term Task Force (NTTF) recommendations are necessary to support issuance of the Vogtle combined licenses, the staff agrees that such conditions may be viable regulatory tools. The NTTF recommendations relevant to COL applications are directed to a relatively narrow set of technical issues, which are not already addressed within the scope of the AP1000 design. The relevant NTTF recommendations relate to enhancing onsite emergency response capability and emergency planning. Accordingly, any resulting conditions would be focused on these particular considerations. However, for reasons explained below, including the Commission's precedent regarding the appropriate use of license conditions, and consistent with the information provided in SECY-11-0137 (Oct. 3, 2011), the staff does not have sufficient information to propose such conditions at this time. The viability of any specific language would depend on what recommendations obtain Commission approval and how they are to be implemented. Following those determinations, the staff is confident that it could develop specific license conditions responsive to the Commission's instructions.

It is important to note at the outset that the Vogtle application meets all current regulatory requirements, and the staff continues to conclude that the application provides reasonable assurance of adequate protection of the public health and safety. For that reason, the staff concluded that the COLs could be issued, without the need for any new license conditions associated with the Fukushima NTTF recommendations. That is why the staff has acknowledged that the Commission can proceed to authorize issuance of the licenses and use existing regulatory approaches if the Commission's ultimate action to implement some or all of the NTTF recommendations does warrant modification of any issued licenses. This approach would provide adequate mechanisms to address regulatory changes the Commission subsequently determines are necessary. As explained in the staff's SECY information paper [SECY-11-0110, Staff Statement in Support of the Uncontested Hearing for Issuance of Combined Licenses and Limited Work Authorizations for Vogtle Electric Generating Plant (VEGP), Units 3 and 4 (Docket Nos. 52-025 and 52-026)], such future modifications would remain subject to applicable finality provisions under 10 CFR Part 52.

However, as emphasized above, if the Commission's view is that additional steps need to be taken now to support the findings for COL issuance, the staff agrees that regulatory controls could be imposed on the license before issuance, including use of license conditions. That said, the specific language and the legal viability of such conditions is dependent both on the exact recommendations that the Commission would choose to implement, the nature of how the Commission would seek to apply it to the COL applicant, and the basis given for implementing the particular recommendation. Neither of those has been determined at this time.

While Commission precedent does allow for reliance on license conditions, such conditions must be "precisely drawn so that the verification of compliance becomes a largely ministerial

rather than an adjudicatory act." *See Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-00-13, 52 NRC 23, 34 (2000). The Commission has further stated that "the mechanism of post-hearing resolution must not be employed to obviate the basic findings prerequisite to an operating license – including a reasonable assurance that the facility can be operated without endangering the health and safety of the public." *Consolidated Edison Co.* (Indian Point Station, Unit 2), CLI-74-23, 7 AEC 947 (1974). Thus, any license condition must be drafted in such a way that the means of compliance with it can be objectively determined at the time the license is issued. Likewise, any license condition must be drafted such that it could not be interpreted as evidence that the staff does not have reasonable assurance of adequate protection of the public health and safety at the time the COL is issued. In short, a license condition could not simply be a generalized "placeholder" binding the licensee to agree to implemented unspecified future Fukushima-related recommendations. Accordingly, it would be difficult to draft a license condition in the absence of specific guidance from the Commission regarding what NTTF recommendations are to be implemented and what those recommendations would require a licensee to do (or provide).

The Fukushima NTTF specified certain aspects of its recommendations that it indicated would be applicable for near-term COL applications. Furthermore, the staff has provided its input on prioritizing the implementation of these recommendations in SECY-11-0137. These NTTF recommendations applicable to the Vogtle COL are:

- Enhance onsite emergency response capability through the integration of emergency operating procedures, severe accident management guidelines, and extensive damage mitigation guidelines; and
- Enhance emergency planning to address prolonged station blackout and multi-unit accidents.

The ultimate Commission determination on how to implement one of these Fukushima-related recommendations might be, for example, to require a licensee to implement a particular management guideline or operating procedure. With that kind of more detailed and objective instruction, the staff would likely have sufficient information to draft a viable license condition that could be added to a COL now as a prerequisite to issuance. Such a condition could require the specific change or addition to be made by a particular time - for example, prior to fuel load.

As explained in its response to the Commission's prehearing questions, assuming such specific Commission direction regarding the form of such Fukushima-related recommendations, the Staff anticipates that preparing an appropriate combination of license conditions would be a relatively straightforward process. That process would entail information gathering and coordination of technical experts, as well as appropriate communication with the applicant, and would likely take time on the order of weeks.

However, as is evident in the Staff's recent response to the Commission in SECY-11-0137 regarding which Fukushima-related recommendations to prioritize, the specific nature of the enhancements that would result from these recommendations is not yet determined. For example, the staff paper indicated that stakeholder involvement would be an important prerequisite to developing the content of the rulemakings that it recommended the Commission undertake. As stated previously, the staff believes that once the parameters of the recommendations are established, development of a license condition could be relatively straightforward. But without those objective parameters, imposing a broad "placeholder" license condition would not be compatible with the Commission's precedent for license issuance.

Question 2:

What process is the industry using on an ongoing basis to factor operating experience from across the world into construction best practices?

This question was directed solely to the applicant. Accordingly, the staff has not provided a response.

Question 3:

For COL action items for the Vogtle COL, please provide a breakdown of how each was resolved and whether each action was identified in the DCD to be completed by the applicant or holder. For each COL action item to be completed by the holder, identify all associated license conditions, ITAACs, or other requirements that ensure the action will be completed. If a COL action item for the holder is not imposed by a requirement, please explain why the action is not necessary to support the conclusions of the FSER or the required findings.

Staff Response:

Combined License (COL) action items (COL Information Items) are information requirements in the DCD that the COL applicant must provide to comply with the requirements for obtaining a combined license. The attached table (Staff Table 1) provides a summary of how each COL action Item (information item) was addressed. It also provides the status of each item as either "Resolved, "FSAR Commitment", "License condition" or "ITAAC". The term "Resolved" means that the COL information item was completely addressed and does not require any unique regulatory tools to validate or assess compliance. The staff determined that none of these post-licensing commitments defers receipt of information that the staff needs in order to make the findings necessary to support initial COL issuance.

For those COL information items that were determined by the staff to be subject to post-licensing commitments, the staff utilized the guidance provided in ESP/DC/COL-ISG-015, "Interim Staff Guidance on Post-Combined License Commitments," to determine whether the appropriate form of post-COL commitment was an ITAAC, license condition, or FSAR commitment. ISG-015 is a publicly available document and may be accessed at ADAMS Accession No. ML091671355 or at the following link: http://www.nrc.gov/reading-rm/doc-collections/isg/col-app-design-cert.html

Question 4:

Regarding the containment debris limitation, please describe how Tier 2* information in the DCD will be captured to ensure that the appropriate change processes and limits are followed during plant operation. Compare the practical implications of reliance on the regulatory provisions in Appendix D of Part 52 to reliance on technical specifications during plant operation.

Staff Response:

There is little practical difference between reliance on the Tier 2* regulatory provisions in Appendix D of Part 52 for controlling debris limits as compared to reliance on technical specifications during plant operation. Both are requirements on the licensees, and both receive regulatory oversight. Assuming a technical specification had been imposed instead of designating the limits as Tier 2*, an out of tolerance condition would be detected no earlier, the overall corrective action would be the same, and changes would fall under the same change provisions as if a technical specification had been imposed (i.e., a license amendment). One potential difference may be that, in general the control room operators would focus more attention to the technical specifications and the day to day operation of the plant than on other requirements. However, in the case of these debris limits, that is not really possible. Latent debris is not a process variable that is continuously monitored and thus would not benefit from additional control room attention. The general housekeeping or maintenance activities associated with the cleanliness program are better controlled by maintenance personnel through maintenance programs. In sum, the use of technical specifications to set limits on containment debris does not provide a practical advantage over the regulatory control through Tier 2* as established by the staff.

As background information for answering this question, it is important to recognize that the AP1000 has eliminated most sources of debris from the design. For example, the design precludes fibrous insulation from the zone of influence. These design features are described in the DCD and verified through ITAAC. Because the design precludes almost all fiber, the only source of fiber is latent debris. This is debris that is inadvertently brought or left in containment during maintenance and outages. The cleanliness program that is identified in the DCD is intended to control and limit the introduction of debris into containment during containment entries for maintenance. The licensee is responsible for the development and implementation of the detailed procedures governing the containment cleanliness program. The program requires the licensee to control the materials brought into containment and track their removal. Consequently, the sampling included in the program is just a confirmation that the overall program controls have been effective. The main contributors to debris have been eliminated by design and other potential sources (i.e. the latent debris) limits are included in Tier 2*.

If, through the testing included in the cleanliness program, the licensee discovers samples that are out of tolerance (i.e., above the limit), the licensee would have to enter the issue into its corrective action program. An out of tolerance sample would represent a condition adverse to quality for a safety-related parameter and Appendix B of Part 50 would therefore require action by the licensee. Criterion XVI of Appendix B, “Corrective Action,” specifically requires that nonconformances be promptly identified and corrected. Appendix B addresses the full range of activities that affect the safety-related functions of SSCs, including how they are designed, purchased, fabricated, handled, shipped, stored, cleaned, erected, installed, inspected, tested, operated, maintained, repaired and modified. Implementation of the corrective active program is

a requirement and it is therefore subject to routine NRC inspection under the reactor oversight program.

If the containment cleanliness is found outside of the design basis assumptions, the licensee would either have to clean the containment and bring it into tolerance and perform another sample or, if justifiable, request a license amendment from the NRC to raise the debris limits designated as Tier 2*, as allowed by Appendix D of Part 52, paragraph VIII.B.6.b.

A situation where the licensee found itself outside the design basis occurred at an operating reactor in the past year in the United States. In this case, the licensee discovered that the GSI-191 debris source term analysis was inconsistent with the as-found plant condition. Accordingly, the reactor was shut down and the situation was corrected before the plant restarted.

Out of tolerance quantities of latent debris could potentially degrade the performance of the passive core cooling systems (PXS), which is addressed in the technical specifications. As a result, in addition to promptly correcting the condition adverse to quality in accordance with the Appendix B requirements, the licensee would also have to promptly assess PXS operability (see GL 91-18 and RIS 2005-20, Rev 1). If such an assessment finds that the PXS is inoperable, the licensee would have to take the remedial actions required by the technical specifications. In addition to entering the issue into its corrective action program and assessing PXS operability, the licensee would also have to evaluate the issue for reportability and evaluate ways to prevent recurrence. Thus, even specifying debris limits solely in Tier 2* does not remove the role of TS in determining the associated operator actions. Again, all of these activities would be subject to NRC inspection.

Question 5:

Squib Valve Surveillance Requirements:

a. Tr. at 160–64
Staff explained during the hearing that the current ASME Code relevant to the squib valves is not adequate and is currently under revision. Therefore, the FSAR states that the surveillance requirements developed for the valves must incorporate lessons from that ongoing process. Staff further stated that the adequacy of those requirements will be determined during the operational program inspection before fuel load. Please explain the relevance of the findings of that inspection and any agency decision concerning operation of the plant including the regulatory basis for any agency action under 10 C.F.R. § 52.103.

b. Tr. at 175
The Staff stated during the hearing that, while it has sufficient experience to describe a surveillance test for squib valves now, it did not do so because it wanted to exercise flexibility next year. Please explain the reasons for omitting a description of the necessary tests prior to the issuance of the COL to preserve flexibility and the basis for concluding that, without a description of the test, this complies with 10 C.F.R. § 52.97(b).

Staff Response:

a. Staff Response:

In its review of the Vogtle Units 3 and 4 COL application, the NRC staff followed the process specified in Commission Paper SECY-05-0197 for the evaluation of operational programs to be developed by a COL applicant. Under that process, the NRC staff reviews a description of the operational programs provided by the COL applicant without establishment of ITAAC in reaching a reasonable assurance finding on the COL application. Under the SECY-05-0197 process, the staff must be satisfied at the COL application stage that the description of the operational program is sufficient to support that reasonable assurance finding. Following COL issuance, the staff will conduct inspections of the operational programs to verify that the programs implement the provisions in the FSAR prior to plant operation.

As discussed in SRM-SECY-02-0067 on COL operational programs, if the Commission determines prior to operations that a licensee will not be in compliance with a regulation or a portion of the license, the normal enforcement process still applies. SRM-SECY-02-0067 also noted that if the Commission finds that the licensee's programs do not provide adequate protection of public health and safety, the staff would take appropriate enforcement action to prohibit or delay fuel load pending appropriate corrective action.

To ensure that the Commission is aware of the status of the operational program implementation at the time of the 10 CFR 52.103(g) finding, Inspection Manual Chapter 2504, "Construction Inspection Program- Inspection of Construction and Operational Programs," included provisions for Commission notification. Specifically, IMC-2504 section 08.02.e states the following:

> *Confirmation of Operational Programs – The staff will inform the Commission of the status of these programs before the Commission makes the determination that the licensee can load fuel. The report to the Commission will convey whether the status of the operational programs is consistent with applicable regulations, license conditions, licensee commitments, and/or the FSAR. As discussed further in Section 09.02 below, it is recognized that some operational programs may not be fully implemented at the time of the 10 CFR 52.103(g) finding.*

Additionally, in SRM-SECY-04-0032, the Commission stated for COL operational programs that the staff should continue the practice of inspecting relevant licensee procedures and programs in a similar manner as was done in the past and consistent with applicable inspection programs. The Commission also noted in SRM-SECY-04-0032 that the staff should continue to ensure, consistent with the inspection and enforcement processes, that licensees address pertinent issues prior to fuel loading. As discussed below, the staff's evaluation of the squib valve inservice testing (IST) program description in the Vogtle COL application is being conducted consistent with this approach for the review, implementation, and inspection of operational programs.

In Vogtle FSAR Section 3.9.6, "Inservice Testing of Pumps and Valves," the COL applicant describes the IST operational program to be developed at Vogtle Units 3 and 4 through incorporation by reference of the provisions in AP1000 DCD Tier 2 Section 3.9.6 and supplemental plant-specific provisions. The description of the IST operational program in the Vogtle FSAR is based on the ASME *Code for Operation and Maintenance of Nuclear*

Power Plants (OM) Code, 2001 Edition through the 2003 Addenda, which includes provisions for IST surveillance of explosive-actuated valves that were developed for current operating plants. In addition, the Vogtle FSAR specifies that the IST program for squib valves will incorporate lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of squib valves to perform their safety functions. Based on the requirements in 10 CFR 50.55a to implement the ASME OM Code and on the IST program description in the Vogtle FSAR, the NRC staff found that the Vogtle COL application adequately describes the IST program for squib valves for incorporating the lessons learned from the design and qualification process, such that there is reasonable assurance of the operational readiness of squib valves to perform their safety functions that supports COL issuance.

The staff will conduct inspections of the Vogtle IST operational program prior to plant operation to verify that surveillance activities for squib valves incorporate lessons learned from the design and qualification process. The staff will base its evaluation of the IST program for squib valves on the NRC regulations in 10 CFR 50.55a that will incorporate by reference the ASME OM Code edition 12 months before fuel load, including squib valve surveillance requirements with any modifications specified in 10 CFR 50.55a that reflect lessons learned from the squib valve design and qualification process. As indicated by the activities discussed in response to Question 5.b below, the staff has confidence that by the time of this milestone for Vogtle Units 3 and 4, the OM Code will incorporate, either directly or through conditions specified in 10 CFR 50.55a when the Code edition is incorporated by reference in the regulations, squib valve surveillance requirements that reflect those lessons learned. However, if the NRC staff finds that the Vogtle IST program has not satisfied the regulatory requirements in 10 CFR 50.55a as well as the FSAR provisions, the staff could take enforcement action to prohibit or delay fuel load pending appropriate corrective action.

While 10 CFR 52.103 applies to authorizing facility operation based on all ITAAC being met, the NRC regulations in 10 CFR 52.98 provide a means for the Commission to delay or prohibit plant start-up where the IST operational program (or any operational program) is found not to comply with the licensing basis for Vogtle Units 3 and 4. If the NRC staff finds during its IST operational program inspections that the COL licensee has not adequately incorporated the lessons learned from the design and qualification process in its IST program for squib valves, to comply with both the applicable 10 CFR 50.55a requirements and the applicant's FSAR commitment, the Commission may require modification of the IST program procedures to meet the FSAR commitment in accordance with the compliance backfit provisions in 10 CFR 50.109(a)(4)(i). Therefore, the staff is confident that Vogtle Units 3 and 4 will not start up prior to the implementation of IST surveillance activities that provide reasonable assurance of the operational readiness of the squib valves to perform their safety functions.

b. Staff Response:

In Vogtle FSAR Section 3.9.6, the COL applicant describes the IST operational program to be developed at Vogtle Units 3 and 4 through incorporation by reference of the provisions in AP1000 DCD Tier 2 Section 3.9.6 and supplemental plant-specific provisions. The description of the IST operational program in the Vogtle FSAR is based on the ASME OM Code, 2001 Edition through the 2003 Addenda, which includes provisions for IST surveillance of explosive-actuated valves for current operating plants. In addition, the

Vogtle FSAR specifies that the IST program for squib valves will incorporate lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of squib valves to perform their safety functions. Based on the requirements in 10 CFR 50.55a to implement the ASME OM Code and the IST program description in the Vogtle FSAR, the NRC staff found that the Vogtle COL application adequately describes the IST program for squib valves for incorporating the lessons learned from the design and qualification process, such that there is reasonable assurance of the operational readiness of squib valves to perform their safety functions that supports COL issuance.

The NRC staff considered the IST program description for squib valves, including the applicant's commitments to incorporate design and qualification lessons learned into the program, to be adequate for COL issuance. Thus, while this approach fundamentally ensures the adequacy of the program by requiring that the IST program incorporate lessons learned from the squib valve design and qualification process, it also allows some flexibility in establishing specific provisions from among the viable surveillance methods examined in that process. The description of the IST program for squib valves in the Vogtle FSAR is information that may be modified by the licensee using the 10 CFR 50.59 process in accordance with 10 CFR 52.98(c)(2). However, the staff expects that a change to the description of the IST program for squib valves would require a license amendment by the COL licensee because such a change would satisfy several of the criteria in 10 CFR 50.59(c)(2) [e.g., more than minimal increase in the likelihood of occurrence of component malfunction].

Furthermore, several factors provide added assurance that the IST program for squib valves will be developed consistent with the description of the program in the Vogtle FSAR:

The NRC regulations in 10 CFR 50.55a(f)(4)(i) specify that COL licensees must implement the edition and addendum of the ASME OM Code that is incorporated by reference in 10 CFR 50.55a 12 months before fuel loading. The staff is preparing a proposed rule to incorporate by reference in 10 CFR 50.55a the 2011 Addenda to the ASME OM Code. The 2011 Addenda to the ASME OM Code includes several improvements to the IST requirements for new reactors, but does not provide improved squib valve surveillance provisions. The staff is preparing a proposed rule to impose conditions on the incorporation by reference of the 2011 Addenda to the ASME OM Code that would specify squib valve surveillance requirements for new reactors based on lessons learned to date from the squib valve design and qualification process. For example, additional squib valve surveillance requirements being considered for the proposed rule include a requirement that new reactor licensees conduct visual external inspections and non-destructive internal inspections (including identification and removal of moisture that interferes with operation of the valve) of squib valves at least once every 2 years. The inspections would be required to be sufficient to verify the operational readiness of the valve and its actuator. At least once every 10 years, the licensee would be required to remove and disassemble each squib valve for internal inspection of the valve and actuator in order to verify the structural integrity of individual components and to remove any foreign material. The staff plans to issue the proposed rule for public comment early in 2012.

Meanwhile, ASME is preparing a revision to the ASME OM Code to provide additional improvements to IST requirements for new reactors, including improved squib valve surveillance provisions. The NRC staff is participating in this ASME Code activity. The

staff will initiate rulemaking to incorporate by reference future editions of the ASME OM Code with any appropriate conditions. If the ASME OM Code is revised to include acceptable IST provisions for squib valves for new reactors, the staff could remove any conditions related to squib valve surveillance requirements specified in the regulations as part of the incorporation by reference of earlier ASME OM Code editions or addenda. In accordance with 10 CFR 50.55a, the COL licensee would be required to implement the edition and addendum of the ASME OM Code incorporated by reference 12 months before fuel loading to ensure that the most recent IST requirements for squib valves are implemented prior to plant startup.

The NRC staff is continuing to monitor the design and qualification process for the squib valves to be used at Vogtle Units 3 and 4, including the development of surveillance requirements for testing and internal inspection. In addition, the staff is planning to conduct a vendor inspection to evaluate the design and qualification process for the squib valves at the Copes-Vulcan facility in Erie, Pa. The staff will also conduct ITAAC inspections to confirm that the squib valves are qualified to perform their safety functions as part of the ITAAC closure process prior to plant startup.

The NRC staff will conduct inspections of the Vogtle IST operational program prior to plant operation to verify that surveillance activities for squib valves incorporate lessons learned from the design and qualification process. The staff will base its evaluation of the IST program for squib valves at Vogtle Units 3 and 4 on the NRC regulations in 10 CFR 50.55a, and the Vogtle FSAR commitment that the IST program for squib valves will incorporate lessons learned from the design and qualification process for these valves such that surveillance activities provide reasonable assurance of the operational readiness of squib valves to perform their safety functions. Therefore, while the staff has confidence at this time that the relevant requirements will be prescribed by the rulemaking, the Vogtle FSAR commitment provides sufficient regulatory control to ensure that the IST program for squib valves will provide reasonable assurance even if the rulemaking is still in progress. If the staff finds that the IST program for squib valves is not in compliance with the licensing basis for Vogtle Units 3 and 4, the staff would apply the enforcement process that could include prohibiting plant startup until corrective action is completed. Again, the staff considers this to be consistent with the level of information provided for operational programs to support findings for COL issuance.

In its SER, the NRC staff concluded that there was reasonable assurance to support COL issuance for Vogtle Units 3 and 4 with respect to the IST program for squib valves because the NRC regulations in 10 CFR 50.55a require the application of the ASME OM Code, and the Vogtle FSAR specifies that the IST program for squib valves will incorporate the lessons learned from the design and qualification process for the operational readiness of squib valves to perform their safety functions. The staff found this approach to be consistent with the Commission policy established in SECY-05-0197. As noted above, the ASME OM Code incorporated by reference in 10 CFR 50.55a includes IST requirements for squib valves that are being implemented at current nuclear power plants. Those surveillance requirements will be improved based on lessons learned from the design and qualification process for squib valves to be used in new reactors through the current rulemaking process underway to incorporate by reference the recent ASME OM Code edition and addenda with planned conditions for squib valve surveillance requirements, and the incorporation by reference of the future edition of the OM Code that is being developed by ASME to provide improved IST requirements for squib valves in new reactors with appropriate conditions specified in 10 CFR 50.55a.

The NRC staff reached a reasonable assurance finding based on its review of the COL application for Vogtle Units 3 and 4 and, for the reasons above, considers the existing analysis and FSAR commitment to be sufficient. However, if the Commission determines that it is necessary and appropriate in reaching its decision, a license condition could impose specific squib valve surveillance requirements for Vogtle Units 3 and 4 based on lessons learned to date from the squib valve design and qualification process. Such a license condition could be constructed to be superseded when the ASME OM Code is updated to include squib valve surveillance provisions for new reactors and is incorporated by reference in the regulations with any appropriate conditions.

Question 6:

The COL application is required to include emergency plans that comply with Appendix E to Part 50. 10 C.F.R § 52.79(a)(21). Part 50, Appendix E, provides, in B "Assessment Actions," that initial EALs must be described, agreed upon by the Applicant and state and local government officials, and approved by the NRC. From the discussion during the hearing, it appears that these requirements have not been satisfied. Instead, the Staff stated it reviewed and approved a plan for developing EALs. Please respond to the following questions:

a. Since the regulation requires NRC approval of the initial EALs, was the Applicant granted an exemption of the requirement to describe the EALS that are to be used? If not, why not?

b. Are there any other instances where the Staff accepted a plan in lieu of any of the application contents required under 10 C.F.R § 52.79(a)(21)?

c. The EAL license condition is silent on whether the NRC review and approval is required. Does the Staff plan to review the submittal?

<u>Staff Response:</u>

a. The applicant was not granted an exemption from the requirement to describe the EALs that are to be used. An exemption for the Vogtle EAL scheme was not needed because Vogtle provided sufficient information to permit the Staff to make a finding of reasonable assurance that Vogtle will meet the applicable requirements when the COL is issued. 10 C.F.R. § 52.79(a)(21) requires COL applicants to submit emergency plans that comply with 10 C.F.R. § 50.47 and Part 50 Appendix E, and § 50.47(b)(4) requires that the emergency plans have a standard emergency classification and action level scheme (referred to as the "EAL scheme"). The EAL scheme consists of the overall program for how emergencies are recognized and classified.

   Vogtle provided an overview of the EAL scheme, including defining its four emergency classification levels. In addition, Vogtle committed to follow, and proposed a license condition requiring it to follow, NEI 07-01, "Methodology for Development of Emergency Action Levels – Advanced Passive Light Water Reactors," Revision 0, with no deviations. NEI 07-01 (Revision 0) is an NRC-approved document for developing EALs for an AP1000, and provides specific guidance on how the EALs will be developed once all necessary as-built, site-specific information is available. By providing an overview of the EAL scheme,

and committing to submit a fully developed set of plant-specific EALs that follow NEI 07-01 (Revision 0), Vogtle has provided its EAL scheme in sufficient detail for the Staff to find that the emergency plan meets the requirements in § 50.47(b)(4) and Appendix E. Therefore, Vogtle has provided an acceptable EAL scheme sufficient to issue the COL.

The Staff will have further verification that the EALs have been properly updated because ITAAC 1.1.2 requires an analysis to be performed of the EAL technical bases to verify as-built, site-specific implementation of the EAL scheme. In addition, ITAAC 8.1 requires a full participation exercise prior to fuel load that will demonstrate the use and adequacy of the EAL scheme for both the licensee and State and local officials.

b. No. As explained above, with respect to EALs the staff did not accept a plan "in lieu of any of the application contents" because the application did comply with 52.79(a)(21). The staff is not aware of <u>any</u> instances where the Staff accepted a plan in lieu of any of the application contents required under 10 C.F.R § 52.79(a)(21).

c. Consistent with its approach to determining compliance with other license conditions, the Staff will confirm that the fully developed EAL scheme was developed in accordance with NEI 07-01, Revision 0, with no deviations when it is submitted by the licensee.

Question 7:

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?

This question was directed solely to the applicant. Accordingly, the staff has not provided a response.

Question 8:

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?

This question was directed solely to the applicant. Accordingly, the staff has not provided a response.

Question 9:

In April 2011, the DNFSB (Defense and Nuclear Facilities Safety Board) 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[1] 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?

<u>Staff Response:</u>

Yes, the applicant, Southern Nuclear Company (SNC), utilized the software program SASSI for performing site-specific soil-structure interaction analysis. Although the SASSI issue is relevant to the Vogtle three-dimensional (3D) soil-structure interaction analysis, staff finds the issue to have no impact on the computed responses of the nuclear island seismic demands, as currently documented in the Vogtle FSAR. Accordingly, the DNFSBs concerns do not have safety significance for the Vogtle building design. The basis for this finding is discussed below.

The staff is aware of the SASSI anomaly described in the April 2011 DNFSB letter and has had follow-up discussions with the DNFSB to better understand the issue. The staff understands the basis of the issue to center on a user-defined modeling approximation that allows the SASSI analyst, running a 3D problem, to use the Subtraction Method (SM) of analysis which uses many fewer degrees of freedom, thereby reducing problem size and run-times. The DNFSB letter identified that this approach can sometimes lead to undefined frequencies of vibration in the problem and can produce spurious results. The anomaly can occur in 3D analyses of embedded structures and is not an issue for analyses using the Direct Method (DM). In addition, the most significant contributing factors for spurious results in SASSI models using the SM relate to the site-specific parameters, including foundation soil characteristics.

For the Vogtle COL application, SNC made use of an embedded 3D SASSI analysis model (i.e., NI-15 model) and the SM approach. This analysis is therefore susceptible to the issue raised by the DNFSB.

To assess the safety significance of the DNFSB issue for the Vogtle site, staff evaluated the analysis results submitted by SNC in the COL application as follows:

---

[1] *See* http://www.dnfsb.gov/board-activities/reports/staff-issue-reports/issues-relatedsassi-computer-software for a link to the letter.

1. Staff reviewed Vogtle in-structure seismic demands for a broad range of response frequencies and noted generally smooth response spectra at key nuclear island locations. Spurious results indicated by abrupt changes in the response spectra, indicative of the behavior cited in the DNSFB letter, were not observed.

2. The staff compared the Vogtle analysis results to a comparable generic site condition analyzed in the AP1000 DCD, which used the Direct Method, and found the results to have similar response characteristics, thereby providing additional assurance as to the absence of spurious behavior.

3. Staff noted that, in the range of frequencies important to the design of Vogtle structures, systems and components (SSCs), there is a significant margin (more than a factor of two) between the site-specific demands and the AP1000 standard design demands, which govern the design of SSCs.

Based on these considerations, staff finds that the concerns raised by the DNFSB have no safety significance for the Vogtle site and that the staff findings as described in FSER Sections 3.7.1.4 and 3.7.2.4 remain valid.

Question 10:

There is an ITAAC in Table 3.6-1, Pipe Rupture Hazards Analysis ITAAC, which requires that an <u>as-designed</u> pipe rupture hazard analysis report exist. The report must conclude that the analysis performed for high and moderate energy piping confirms the protection of structures, systems, and components required to be functional during and following a design basis event. There is a very similar ITAAC in Table 3.12-1, Piping Design ITAAC, for compliance of the as-designed piping ITAAC, for compliance of the as-designed piping with ASME Code. What commitment, programs, or license conditions do we have in place to assure ourselves that the <u>as-installed</u> piping will match the <u>as-designed</u> piping to ensure that our safety conclusions remain valid?

<u>Staff Response:</u>

There are a total of four ITAAC to address the two areas. As stated in Appendix B of Vogtle COL Application Part 10- License Conditions and ITAAC, the Tier 1 information (including the ITAAC) of the referenced DCD (AP1000) is incorporated by reference with some departures and/or supplements identified. These departures and/or supplements include the as-designed aspect of pipe rupture hazard analysis report (refer to Appendix B Tables 3.8-1) and the <u>as-designed</u> piping analysis (refer to Appendix B Table 3.8-2) and are identified as COL Items 3.6-1 and 3.9-7, respectively. These two site-specific ITAAC are intended to verify that the design follows the methodology specified in the DCD. These two ITAAC and two license conditions intended to address timing are included to address Piping Design Acceptance Criteria.

The <u>as-built</u> (or as installed) reconciliation of the pipe rupture hazard analysis report is addressed through a separate ITAAC in AP1000 DCD Tier 1 ITAAC (non-system based) Table 3.3-6 Line Item 8. Similarly, the as-built reconciliation of piping design is also addressed through a set of separate ITAAC in AP1000 DCD Tier 1 Section 2 (System based ITAAC Tables) line 2.b. These ITAAC are incorporated by reference in the Vogtle application. The

intent of these ITAAC is to ensure that changes in the design are reconciled to verify that the as-built design continues to be in accordance with the ASME Code and the regulations.

Question 11:

In Chapter 3, FSER section 3.7.1.4 the Staff states that the Applicant performed site-specific SSI analysis and there are a limited number of locations in the structure where the AP1000 design is exceeded. The Staff states the impacts of these exceedances have been evaluated and the justification provided by the Applicant ensures the design has not been compromised. What did the Staff do to justify this conclusion?

Staff Response:

To address site-specific exceedances of the AP1000 certified seismic design response spectra (CSDRS) by the VEGP ground motion response spectra (GMRS), the applicant, Southern Nuclear Company (SNC), performed detailed soil-structure interaction analysis and provided justifications for design exceedances in VEGP FSAR Appendix 3GG.

The staff verified that the VEGP seismic analysis models were developed in accordance with SRP guidance. The staff's evaluation is described in FSER Section 3.7.2.4. To confirm the SNC justification for the design exceedances, staff reviewed the results of VEGP FSAR Appendix 3GG and AP1000 DCD Appendix 3G and performed several tasks:

(a) Identified the nuclear island locations where the exceedances occur,
(b) Estimated the magnitude of the exceedances,
(c) Compared the frequency range corresponding to the exceedance to the significant resonant frequencies of safety-significant AP1000 structures, systems, and components (SSCs), and
(d) Compared available design margin in the frequency ranges that are significant to the design of SSCs.

The results of the staff's review showed that there were design exceedances at several nuclear island key locations. Further, staff found that the frequency range corresponding to the design exceedances was approximately 0.50-0.60 Hz and that the magnitudes of these exceedances were all less than 10-percent. To assess the significance of these exceedances for the AP1000 standard design, staff performed a review of the resonant frequencies for safety-significant SSCs. Staff found that the resonant frequencies for nuclear island structures were all greater than 2 Hz and that resonant tank sloshing frequencies were all outside of the narrow range of the exceedance. In addition, staff notes that for the frequency ranges that are significant to the design of SSCs (i.e., greater than 2 Hz), the VEGP seismic analysis described in VEGP FSAR Appendix 3GG, show that there is a significant margin (more than a factor of two) between the site-specific demands and the AP1000 standard design demands, which govern the design of SSCs. Consequently, the staff concluded that small exceedance of the VEGP seismic demands over the AP1000 standard design will not negatively affect the design of safety-significant SSCs.

Based on the staff findings that (a) the VEGP seismic analysis was performed in accordance with SRP guidance, (b) the frequency range of the VEGP design exceedance does not correspond to resonant frequencies of safety-significant SSCs, (c) the magnitude of the exceedance is small (less than 10-percent), and (d) significant margin exists in frequency ranges important to design, the staff finds the SNC justification, which showed that the AP1000

design has not been compromised, to be acceptable.

Question 12:

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?

Staff Response:

The staff elected not to require a technical specification for the debris limits, as recommended by the ACRS, for several reasons. First and foremost, the elements of the cleanliness program itself are sufficient to provide assurance of containment cleanliness prior to start-up, without the need for sampling. It is important to note that the staff's evaluation of the AP1000 design concluded that the design complies with the provisions of 10 CFR 50.46. The AP1000 has demonstrated compliance with 10 CFR 50.46 by carefully controlling the amount of debris that could be generated in a loss of coolant accident and by establishing limits on latent debris. The AP1000 has eliminated most sources of debris from the design. For example, the design precludes fibrous insulation from the zone of influence. These design features are described in Tier 1 and Tier 2 of the DCD and verified through ITAAC. Because the design precludes almost all fiber, the only source of fiber is latent debris. This is debris that is inadvertently brought into or left in containment during maintenance and outages. The cleanliness program that is identified in the DCD is intended to control and limit the introduction of debris into containment during containment entries for maintenance. The licensee is responsible for the development and implementation of the detailed procedures governing the containment cleanliness program. This program requires the licensee to control which materials are brought into containment and track their removal. The sampling included in the program is just a confirmation that the overall program controls have been effective. The main contributors to debris have been eliminated by design and the other potential sources (i.e. the latent debris) limits are included in Tier 2*.

The technical specification rule, 10 CFR 50.36, requires safety limits for fission product barriers, limiting safety system settings, limiting conditions of operation, surveillance requirements, and design features, and administrative controls. The Commission's 1993 final policy statement on technical specification improvements and 10 CFR 50.36(c)(2)(ii) provide four criteria for establishing a technical specification limiting condition for operation (LCO) (which is taken to also include appropriate associated remedial actions when the LCO is not met). The four LCO criteria consider instrumentation for detecting abnormal degradation of the reactor coolant pressure boundary; accident analysis initial conditions on process variables, design features, and operating restrictions; accident analysis primary success path structures, systems and components (SSCs) for preventing or mitigating accidents or transients; and SSCs that operational experience or probabilistic risk assessment has shown to be significant to public health and safety. Although judgment can be applied when interpreting these criteria, the general housekeeping requirements associated with latent debris do not meet these criteria. Further, in the interest of maintaining technical specifications standardization, the Staff

concluded that addressing the issue once in the DCD with Tier 2* was preferable to each individual COL applicant proposing a site-specific technical specification. Maintenance and housekeeping are very important and are better addressed in DCD Tier 2 and Tier 2* information.

A technical specification limit on latent debris would neither ensure earlier detection of an out of tolerance condition, nor result in different overall corrective actions. Whether the latent debris tolerances are in the technical specifications or in Tier 2*, any changes in the tolerances would fall under the same change provisions—a license amendment request under 10 CFR 50.90. One practical difference may be that in the day-to-day operation of the plant the control room operators might be expected to focus more attention on requirements in technical specifications than on requirements in other documents, such as DCD Tier 2*. However, in the case of these debris limits, that is not really possible. The quantity of latent debris in containment is not a process variable that is continuously monitored during plant operation and would not benefit from additional control room attention, which technical specifications might afford. Therefore, general housekeeping or maintenance activities associated with the cleanliness program are better controlled by maintenance personnel though maintenance programs.

Furthermore, the decision to use Tier 2* for establishing the latent debris limits was not related to whether the results of the sampling would be evaluated pre-start-up or post-start-up. Rather, the latent debris sampling included in the cleanliness program is just one aspect of the overall program. Sampling is confirmatory in nature and serves to demonstrate the effectiveness of the cleanliness program. This approach is applied to other aspects of nuclear power plant operations. For example, operators are periodically tested, diesel fuel is periodically sampled and pumps and valves are subject to periodic testing. There is no presumption in these examples that the safety function is lost between samples or when the samples are being evaluated. Consistent with this philosophy, the staff determined that evaluation of the debris sampling results would not be necessary prior to start-up. With these considerations, evaluation of test results after start-up was acceptable to the staff.

Question 13:

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?

Staff Response:

The staff does not require the applicant to reconcile or monitor the differences between the Caldon LEFM CheckPlus™ and feedwater flow venture meter measurements during power ascension testing or during normal operation. Both the LEFM and flow venturi are calibrated within their measurement uncertainties. The Caldon LEFM CheckPlus™, an ultrasonic flow meter called a "leading edge flow meter," which has been in use in operating plants to reduce a power calorimetric uncertainty to less than 1 percent, is used as the primary instrument for the feedwater flow measurement in the power calorimetric process. The feedwater venturi, which is a differential pressure element widely used in operating plants with less than 2 percent power uncertainty, can also be used for the feedwater flow measurement. Because of lower measurement uncertainty, the LEFM CheckPlus™ would be used for the power calorimetric process when it is operable. In the event that the LEFM is out-of-service, the plant contingency plans call for the use of feedwater flow venturi for the calorimetric process. Since the use of venturi has a 2 percent power uncertainty compared to 1 percent for the LEFM, the contingency plan specifies that the plant using a 2 percent power uncertainty will derate to compensate for the difference.

The commitment to the periodic calibration of the Caldon LEFM and the contingency plan related to the Caldon LEFM out-of-service is captured in the VEGP COL application. In its October 29, 2010, letter, the applicant discussed the maintenance plans, stating that calibration and maintenance for the Caldon LEFM CheckPlus™ system hardware and instrumentation is performed using procedures based on the Caldon LEFM CheckPlus™ technical manual. The applicant also discussed the contingency plans in the event the Caldon LEFM is out of service. The applicant stated that "Plant instrumentation that affects the power calorimetric, including the Caldon LEFM CheckPlus inputs, is monitored by plant system engineering personnel. These instruments are included in the plant preventative maintenance (PM) program for periodic calibration. Problems that are detected are documented per the plant corrective action process and necessary resolution actions are planned and implemented. Corrective action procedures, which provide compliance with the requirements of 10 CFR Part 50, Appendix B, include instructions for notification of deficiencies and error reporting. In addition, an administrative control is provided for the operation of the plant at power with the device out-of-service. The controls provide for de-rating the plant output to power levels consistent with a 2% plant uncertainty…." The staff notes that the use of administrative controls for the maintenance and contingency plans is consistent with operating plants using the Caldon LEFM CheckPlus™ feedwater flow measurement instrument.

Vogtle FSAR Section 15.0.3.2 states that administrative controls implement maintenance and contingency activities related to the power calorimetric instrument. The VEGP license includes a license condition that requires the availability of administrative controls to implement maintenance and contingency activities related to the power calorimetric uncertainty instrumentation, prior to fuel load. The maintenance activities would include periodic calibration of the power calorimetric instrumentation as part of the PM program.

Question 14:

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.

This question was directed solely to the applicant. Accordingly, the staff has not provided a response.

Staff Table 1

ITEM #	SUBJECT	Applicant's FSAR Section	Staff's FSER Section (s)	Status
1.1-1	Construction and Startup Schedule	1.1.5, 1.1.7	1.4	FSAR Commitment
1.9-1	Regulatory Guide Conformance	1.9.1, 1.9.1.1, 1.9.1.2, 1.9.1.3, 1.9.1.4, 1.9.1.5, Appendix 1A, Appendix 1AA	1.4	Resolved
1.9-2	Bulletins and Generic Letters	1.9.5.5	1.4	Resolved
1.9-3	Unresolved Safety Issues and Generic Safety Issues	1.9.4.1, 1.9.4.2.3	1.4	Resolved
2.1-1	Geography and Demography	1.1.1, 1.2.2, 2.1.4	2.1	Resolved
2.2-1	Identification of Site-specific Potential Hazards	2.2.3.2.3.1, 2.2.3.2.3.2, 2.2.3.3, 2.2.3.4, 2.2.4	2.2.3	Resolved
2.3-1	Regional Climatology	2.3.6.1	2.3.1	Resolved
2.3-2	Local meteorology	2.3.6.2	2.3.2	Resolved
2.3-3	Onsite meteorological measurements program	2.3.3.4, 2.3.6.3	2.3.3	Resolved
2.3-4	Short-Term Diffusion Estimates	2.3.4, 2.3.6.4, 15.6.5.3.7.3, 15A.3.3	2.3.4	Resolved
2.3-5	Long-Term Diffusion Estimates	2.3.5, 2.3.6.5	2.3.5	Resolved
2.4-1	Hydrological Description	2.4.15.1	2.4.2	Resolved
2.4-2	Floods	2.4.2, 2.4.10, 2.4.15.2	2.4.2, 2.4.10	Resolved
2.4-3	Cooling Water Supply	2.4.15.3	2.4.12	Resolved
2.4-4	Groundwater	2.4.15.4	2.4.12	Resolved
2.4-5	Accidental Release of Liquid Effluents into Ground and Surface Water	2.4.15.5	2.4.13	Resolved
2.4-6	Flood Protection Emergency Operation Procedures	2.4.14, 2.4.15.6	2.4.14	Resolved

2.5-1	Basic Geologic and Seismic Information	2.5.7.1	2.5.1	Resolved
2.5-2	Site Seismic and Tectonic Characteristics Information	2.5.7.2	2.5.2	Resolved
2.5-3	Geoscience Parameters	2.5.7.3	2.5.2	Resolved
2.5-4	Surface Faulting	2.5.7.4	2.5.3	Resolved
2.5-5	Site and Structures	2.5.7.5	2.5.4	Resolved
2.5-6	Properties of Underlying Materials	2.5.7.6	2.5.4	LC 2-1 and ITAAC Table 2.5-1
2.5-7	Excavation and Backfill	2.5.7.7	2.5.4	LC 2-1 and ITAAC Table 2.5-1
2.5-8	Ground Water Conditions	2.5.7.8	2.5.4	Resolved
2.5-9	Liquefaction Potential	2.5.7.9	2.5.4	Resolved
2.5-10	Bearing Capacity	2.5.7.10	2.5.4	Resolved
2.5-11	Earth Pressures	2.5.7.11	2.5.4	Resolved
2.5-12	Static and Dynamic Stability of Facilities	2.5.7.12	2.5.4	Resolved
2.5-13	Subsurface Instrumentation	2.5.7.13	2.5.4	Resolved
2.5-14	Stability of Slopes	2.5.7.14	2.5.4	Resolved
2.5-15	Embankments and Dams	2.5.7.15	2.5.4	Resolved
2.5-16	Settlement of Nuclear Island	2.5.7.16	2.5.4	Resolved
2.5-17	Waterproofing System	2.5.7.17, 3.4.1.1.1.1, 3.8.5.1	2.5.4, 3.8.5	ITAAC Table 3.8-1
3.3-1	Wind and Tornado Site Interface Criteria	1.2.2, 3.3.1.1, 3.3.2.1, 3.3.2.3, 3.3.3, 3.5.1.5, 3.5.1.6	3.3.1, 3.3.2, 3.5.1	Resolved
3.4-1	Site-specific Flooding Hazards Protective Measures	3.4.1.3, 3.4.3	3.4.1	Resolved
3.5-1	External Missile Protection Requirements	1.2.2, 3.3.1.1, 3.3.2.1, 3.3.2.3, 3.5.1.5, 3.5.1.6, 3.5.4	3.5.1	Resolved
3.6-1	Pipe Break Hazards Analysis	3.6.4.1, 14.3.3.2	3.6.1	LC 3-1, ITAAC Table 3.6-1

3.6-4	Primary System Inspection Program for Leak-Before-Break Piping	3.6.4.4	3.6.1	Resolved
3.7-1	Seismic Analysis of Dams	3.7.2.12, 3.7.5.1	3.7.4	Resolved
3.7-2	Post- Earthquake Procedures	3.7.4.4, 3.7.5.2	3.7.4	Resolved
3.7-3	Seismic Interaction Review	3.7.5.3	3.7.3	LC 3-2
3.7-4	Reconciliation of Seismic Analyses of Nuclear Island Structures	3.7.5.4	3.7.3	LC 3-3
3.7-5	Location of Free-Field Acceleration Sensor	3.7.4.2.1, 3.7.5.5	3.7.5	Resolved
3.8-5	Structures Inspection Program	3.8.3.7, 3.8.4.7, 3.8.5.7, 3.8.6.5, 17.6	3.8.5	Resolved
3.8-6	Construction Procedures Program	3.8.6.6	3.8.5	LC 3-4
3.9-2	Design Specification and Reports	3.9.8.2	3.9.2, 3.12	Resolved
3.9-3	Snubber Operability Testing	3.9.3.4.4, 3.9.8.3	3.9.2	Resolved
3.9-4	Valve Inservice Testing	3.9.6, 3.9.6.2.2, 3.9.6.2.4, 3.9.6.2.5, 3.9.6.3, 3.9.8.4	3.9.6	LC 3-5 and 3-6
3.9-5	Surge Line Thermal Monitoring	3.9.3.1.2, 3.9.8.5, 14.2.9.2.22	3.9.2, 3.12	Resolved
3.9-7	As-Designed Piping Analysis	3.9.8.7, 14.3.3.3	3.9.2, 3.12	LC-3-9, ITAAC Table 3.12-1
3.11-1	Equipment Qualification File	3.11.5	3.11	LC 3-7 and 3-8
4.4-2	Confirm Assumptions for Safety Analyses DNBR Limits	4.4.7	chapter 4	LC 4-1
5.2-1	ASME Code and Addenda	5.2.1.1, 5.2.6.1	5.2.1	Resolved
5.2-2	Plant Specific Inspection Program	5.2.4, 5.2.4.1, 5.2.4.3.1, 5.2.4.3.2, 5.2.4.4, 5.2.4.5, 5.2.4.6, 5.2.4.8,5.2.4.9,5.2.4.10, 5.2.6.2	5.2.2	LC 5-1

5.2-3	Response to Unidentified Reactor Coolant System Leakage Inside Containment	5.2.6.3, 5.2.5.3.5	5.2.5	FSAR commitment
5.3-1	Reactor Vessel Pressure -- Temperature Limit Curves	5.3.6.1	5.3.3	LC- 5-4
5.3-2	Reactor Vessel Materials Surveillance Program	5.3.2.6, 5.3.2.6.3, 5.3.6.2	5.3.2	LC 5-2, 5-3
5.3-4	Reactor Vessel Materials Properties Verification	5.3.6.4.1	5.3.4	LC 5-5
5.3-7	Quickloc Weld Build-up ISI	5.2.4.1, 5.3.6.6	5.2.2	LC 5-1
5.4-1	Steam Generator Tube Integrity	5.4.2.5, 5.4.15	5.4	Resolved
6.1-1	Procedure Review for Austenitic Stainless Steels	6.1.1.2, 6.1.3.1	6.1.1	Resolved
6.1-2	Coating Program	6.1.2.1.6, 6.1.3.2	6.1.2	Resolved
6.2-1	Containment Leak Rate Testing	6.2.5.1, 6.2.5.2.2, 6.2.6	6.2	LC 6-1, 6-2
6.3-1	Containment Cleanliness Program	6.3.8.1	6.3	Resolved
6.4-1	Local Hazardous Gas Services and Monitoring	2.2.3.2.3.1, 2.2.3.2.3.2, 2.2.3.3, 6.4.4.2, 6.4.7	6.4	Resolved
6.4-2	Procedures for Training for Control Room Habitability	6.4.3, 6.4.7	6.4	FSAR commitment
6.6-1	Inspection Programs	6.6, 6.6.1, 6.6.3.1, 6.6.3.2, 6.6.3.3, 6.6.4, 6.6.6, 6.6.9.1	6.6	LC 6-3
6.6-2	Construction Activities	6.6.2, 6.6.9.2	6.6	Resolved
7.1-1	Setpoint Calculations for Protective Functions	7.1.6.1	7.1	Resolved
7.5-1	Post Accident Monitoring	7.5.2,7.5.3.5, 7.5.5	7.5	Resolved
8.2-1	Offsite Electrical Power	8.2.1, 8.2.1.1, 8.2.1.2, 8.2.1.3, 8.2.1.4, 8.2.5	8.2	Resolved
8.2-2	Technical Interfaces	8.2.1.2.1, 8.2.2, 8.2.5	8.2	Resolved

8.3-1	Grounding and Lightning Protection	8.3.1.1.7 8.3.1.1.8, 8.3.3	8.3.1	Resolved
8.3-2	Onsite Electrical Power Plant Procedures	8.3.1.1.2.4, 8.3.1.1.6, 8.3.2.1.4, 8.3.3	8.3.1	Resolved
9.1-5	Inservice Inspection Program of Cranes	9.1.4.4, 9.1.5.4, 9.1.6	9.1.4, 9.1.5	FSAR commitment
9.1-6	Radiation Monitor	9.1.4.3.8, 9.1.5.3, 9.1.6	9.1.4, 9.1.5	Resolved
9.1-7	Metamic Monitoring Program	9.1.6	9.1.2	LC 9-1
9.2-1	Potable Water	9.2.5.2.1, 9.2.5.2.2, 9.2.5.3, 9.2.5.6, 9.2.12.1	9.2.5	Resolved
9.2-2	Waste Water Retention Basins	9.2.9.2.1, 9.2.9.2.2, 9.2.9.5, 9.2.12.2	9.2.9	Resolved
9.3-1	Air Systems (NUREG-0933 Issue 43)	9.3.7	9.3.1	Resolved
9.4-1	Ventilation Systems Operations	6.4.4.2, 9.4.1.4, 9.4.7.4, 9.4.12	9.4.1, 9.4.7	Resolved
9.5-1	Qualification Requirements for Fire Protection Program	9.5.1.6, 9.5.1.8, 9.5.1.8.1.2, 9.5.1.8.2, 9.5.1.8.3, 9.5.1.8.4, 9.5.1.8.5, 9.5.1.8.6, 9.5.1.8.7, 9.5.1.9.1, 13.1.1.2.10	9.5.1	LC 9-2, LC 9-3
9.5-2	Fire Protection Analysis Information	9.5.1.9.2, 9A.3.3	9.5.1	Resolved
9.5-3	Regulatory Conformance	9.5.1.8.1.1, 9.5.1.8.8, 9.5.1.8.9, 9.5.1.9.3, 9A.3.3	9.5.1	Resolved
9.5-4	NFPA Exceptions	9.5.1.8.1.1, 9.5.1.9.4	9.5.1	Resolved
9.5-6	Verification of Field Installed Fire Barriers	9.5.1.8.6, 9.5.1.9.6	9.5.1	Resolved
9.5-8	Establishment of Procedures to Minimize Risk for Fire Areas Breached During Maintenance	9.5.1.8.1.2, 9.5.1.9.7	9.5.1	Resolved
9.5-9	Offsite Interfaces	9.5.2.5.1	9.5.2	Resolved
9.5-10	Emergency Offsite Communications	9.5.2.5.2	9.5.2	Resolved
9.5-11	Security Communications	9.5.2.5.3	9.5.2	Resolved
9.5-13	Fuel Degradation Protection	9.5.4.5.2, 9.5.4.7.2	9.5.4	Resolved
10.1-1	Erosion-Corrosion Monitoring	10.1.3.1, 10.1.3.2, 10.1.3.3	10.1	LC 10-1

10.2-1	Turbine Maintenance and Inspection	10.2.6	10.2	LC 10-2
10.4-1	Circulating Water Supply	10.4.5.2.1, 10.4.5.2.2, 10.4.5.5, 10.4.12.1	10.4.5	Resolved
10.4-2	Condensate, Feedwater and Auxiliary Steam System Chemistry Control	10.4.7.2.1, 10.4.12.2	10.4.7	Resolved
10.4-3	Potable Water	9.2.5.3, 10.4.12.3	9.2.5	Resolved
11.2-1	Liquid Radwaste Processing by Mobile Equipment	11.2.1.2.5.2, 11.2.5.1	11.2	Resolved
11.2-2	Cost benefit Analysis of Population Doses	11.2.3.3, 11.2.3.5, 11.2.5.2	11.2	Resolved
11.3-1	Cost benefit Analysis of Population Doses	11.3.3.4, 11.3.5.1	11.3	Resolved
11.4-1	Solid Waste Management System Process Control Program	11.4.6	11.4	LC11-1,11-2
11.5-1	Plant Offsite Dose Calculation Manual (ODCM)	11.5.8	11.5	LC 11-3, 11-4
11.5-2	Effluent Monitoring and Sampling	11.5.1.2, 11.5.2.4, 11.5.3, 11.5.4, 11.5.4.1, 11.5.4.2, 11.5.6.5, 11.5.8	11.5	LC 11-3, 11-4
11.5-3	10 CFR 50, Appendix I	11.2.3.5, 11.3.3.4, 11.5.8	11.5	Resolved
12.1-1	ALARA and Operational Policies	12.1, 12.1.3, Appendix 12AA	12.1	LC 12-1, 12-2
12.2-1	Additional Contained Radiation Sources	12.2.1.1.10, 12.2.3	12.2	Resolved
12.3-1	Administrative Controls for Radiological Protection	12.3.5.1, Appendix 12AA	12.3	Resolved
12.3-2	Criteria and Methods for Radiological Protection	12.3.4, 12.3.5.2	12.3	Resolved
12.3-3	Groundwater Monitoring Program	12.3.5.3, Appendix 12AA	12.3	Resolved

12.3-4	Record of Operational Events of Interest for Decommissioning	12.3.5.4, Appendix 12AA	12,3	Resolved
12.5-1	Radiological Protection Organization and Procedures	12.5.5, Appendix 12AA	12.5	LC 12-1, 12-2
13.1-1	Organizational Structure of Combined License Applicant	13.1.1, 13.1.2, 13.1.3, 13.1.4, Appendix 13AA	13.1	Resolved
13.2-1	Training Program for Plant Personnel	13.2, 13.2.1	13.2	LC 13-1, 13-2
13.3-1	Emergency Planning and Communications	13.3, 13.3.6, 13.3.7	13.3	LC 13-3, 13-4, ITAAC Tables 13.3-1 and 2
13.3-2	Activation of Emergency Operations Facility	13.3, 13.3.6	13.3	same as above
13.4-1	Operational Review	13.4, 13.4.1	13.4	LC in respective sections of the FSER
13.5-1	Plant Procedures	13.5, 13.5.3	13.5	Resolved
13.6-1	Security Communications	13.6, 13.6.1, 14.3.2.3.2	13.6	LC 13-5
13.6-5	Cyber Security Program	13.6, 13.6.1	13.8	LC 13-7
14.4-1	Organization and Staffing	14.2.2, 14.4.1	14.2.2	Resolved
14.4-2	Test Specifics and Procedures	14.4.2	14.2.3	Resolved
14.4-3	Conduct of Test Program	14.4.3	14.2.1, 14.2.3, 14.2.6	LC 14-1
14.4-4	Review and Evaluation of Test Results	14.2.3.2, 14.4.4	14.4.3	LC 14-2
14.4-5	Testing Interface Requirements	14.2.9.4.15, 14.2.9.4.22, 14.2.9.4.23, 14.2.9.4.24, 14.2.9.4.25, 14.2.9.4.25, 14.2.9.4.26, 14.2.9.4.27, 14.2.10.4.29, 14.4.5	14.2.9, 14.2.10	Resolved
14.4-6	First-Plant-Only and Three-Plant-Only Tests	14.4.6	14.2.5	LC 14-3 and 14-4

15.0-1	Documentation of Plant Calorimetric Uncertainty Methodology	15.0.15, 15.0.3.2	15	LC 15-1 and ITAAC Table 15.0-1
15.7-1	Consequences of Tank Failure	15.7.6	15.7	Resolved
16.1-1	Technical Specification Preliminary Information	16.1.1	16.1	Resolved
16.3-1	Procedure to Control Operability of Investment Protection Systems, Structures and Components	16.3.1, 16.3.2	16.3	Resolved
17.5-1	Quality Assurance Design Phase	17.1, 17.5, 17.7	17.1, 17.5	Resolved
17.5-2	Quality Assurance for Procurement, Fabrication, Installation, Construction and Testing	17.5, 17.7	17.5	Resolved
17.5-4	Quality Assurance Program for Operations	17.5, 17.7	17.5	Resolved
17.5-8	Operational Reliability Assurance Program Integration with Quality Assurance Program	17.7, 17.7	17.4	Resolved
18.2-2	Design of the Emergency Operations Facility	9.5.2.2.5, 18.2.1.3, 18.2.6.2	18.2	Resolved
18.6-1	Plant Staffing	13.1.1.4, 13.1.3.1, 13.1.3.2, 18.6, 18.6.1	18.6	Resolved
18.10-1	Training Program Development	13.1.13.1.3.2.2, 13.2, 18.10, 18.10.1	18.1	Resolved
18.14-1	Human Performance Monitoring	18.14	18.14	Resolved
19.59.10-1	As-Built SSC HCLPF Comparison to Seismic Margin Evaluation	19.59.10.5	19.59	LC 19-1

19.59.10-2	Evaluation of As-Built Plant Versus Design in AP1000 PRA and Site-Specific PRA External Events	19.59.10.5	19.59	LC 19-2
19.59.10-3	Internal Fire and Internal Flood Analyses	19.59.10.5	19.59	LC 19-3
19.59.10-4	Implement Severe Accident Management Guidance	19.59.10.5	19.59	LC 19-4
19.59.10-5	Equipment Survivability	19.59.10.5	19.59	LC 19-5
19.59.10-6	Confirm that the Seismic Margin Assessment analysis is applicable to the COL site	19.55.6.3, 19.59.10.5	19.59	Resolved

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of	)	
	)	
SOUTHERN NUCLEAR OPERATING CO.	)	Docket Nos. 52-025-COL and 52-026-COL
	)	
(Vogtle Electric Generating Plant	)	
Units 3 and 4)	)	
	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC Staff Responses to Commission Post-Hearing Questions" have been served upon the following persons by Electronic Information Exchange this 17th day of October, 2011:

Office of the Secretary
Mail Stop 0-16C1
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
(E-mail: HEARINGDOCKET@nrc.gov)

Office of Commission Appellate
Adjudication
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
(E-mail: ocaamail@nrc.gov)

John L. Pemberton, Esq.
Southern Nuclear Operating Co., Inc.
40 Inverness Center Parkway
P.O. Box 1295, Bin B-022
Birmingham, AL 35201-1295
(E-mail: jlpember@southernco.com)

Balch & Bingham, LLP
M. Stanford Blanton, Esq.
C. Grady Moore, III, Esq.
Millicent Ronnlund
1710 Sixth Avenue North
Birmingham, AL 35203-2014
Phone: 205-251-8100
(E-mail: sblanton@balch.com;
gmoore@balch.com;
mronnlund@balch.com)

Morgan, Lewis & Bockius LLP
Kathryn M. Sutton, Esq.
Mary Freeze
1111 Pennsylvania Avenue, NW
Washington, DC 20004
(E-mail: ksutton@morganlewis.com;
mfreeze@morganlewis.com)

/Signed (electronically) by/
Patrick A. Moulding
Counsel for the NRC Staff
U.S. Nuclear Regulatory Commission
Mail Stop O-15 D21
Washington, DC 20555-0001
(301) 415-2549
Patrick.Moulding@nrc.gov

Dated at Rockville, Maryland
this 17th day of October, 2011