Mr. Gary J. Taylor
Vice President, Nuclear Operations
South Carolina Electric and Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

January 11, 1996

SUBJECT: SUMMER INDIVIDUAL PLANT EXAMINATION RESPONSE TO GENERIC LETTER 88-20, (RC-93-0170) (TAC NO. M74475)

Dear Mr. Taylor:

Your letter of June 18, 1993 submitted the Individual Plant Examination (IPE) for the Virgil C. Summer Nuclear Station for internal events, including internal flooding. To complete our review, we are requesting additional information as identified in the enclosure to this letter. Please provide additional information within 60 days of issuance of this letter. The requirement affects nine or fewer respondents and, therefore, is not subject to the Office of Budget review under P.L. 96-511.

Sincerely,

Original signed by

Jacob I. Zimmerman, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosure: Request for Additional Information

cc w/enclosure: See next page

Distribution
Docket File
PUBLIC
Summer Reading
S. Varga
J. Zwolinski
OGC
ACRS
E. Merschoff

NRC FILE CENTER COPY

KF01 1/1

DOCUMENT NAME: G:\SUMMER\M74475

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PDII-3/LA		DRPE		PDII-3/PM		PDII-3/D	C
NAME	BClayton		RClark		JZimmerman		FHebdon	
DATE	01/3/96		01/03/96		01/11/96		01/11/96	

OFFICIAL RECORD COPY

9601180058 960111
PDR ADOCK 05000395
P PDR

REQUEST FOR ADDITIONAL INFORMATION

INDIVIDUAL PLANT EXAMINATION

VIRGIL C. SUMMER NUCLEAR STATION

DOCKET NO. 50-395

A. Front-End Analysis

1. Please discuss the availability of the turbine-driven emergency feedwater (EFW) pump after battery depletion in a station blackout (SBO). The discussion should include the environmental conditions that may affect the controlling equipment and the operator, where long lasting manual control is assumed to be exercised (e.g., in the EFW room). Explain how emergency feedwater flow is controlled under these conditions. Please discuss the impact of your assumptions regarding EFW pump control on core damage frequency (CDF).

2. At this plant, the SBO contribution to the total CDF is about 21%. It is not clear from the submittal if plant changes due to the SBO rule were credited in the analysis. Please provide the following information:

   a. Report whether plant changes (e.g., procedures for load shedding, AC power) made in response to the SBO rule were credited in the IPE and which specific plant changes were credited.

   b. If available, give the total impact of these plant changes to the total plant CDF and to the station blackout CDF (i.e., reduction in total plant CDF and SBO CDF).

   c. If available, give the impact of each individual plant change to the total plant CDF and the SBO CDF (i.e., reduction in total plant CDF and SBO CDF).

   d. Report any other changes to the plant that are separate from those strictly in response to the SBO rule, that nonetheless may reduce the SBO CDF. In addition, report whether these changes are implemented or planned, report whether credit was taken for these changes in the IPE, and, if available, discuss the impact of these changes to the SBO CDF.

3. It appears that losses of AC buses were not included as initiating events. Please explain the basis for screening these initiators, and provide an estimate on the impact on the CDF and important accident sequences if they were included.

4. A review of the common cause failure (CCF) analysis in the IPE (pages 3-156, 3-157) indicates that some components, in spite of their importance, were not analyzed or are apparently included in "the generic component, All."

a. Provide an explanation of the notion "generic component, All" in deriving generic multiple Greek letter (MGL) parameters for components as diverse as check valves, chillers, or fans.

b. Provide the reasons why the same MGL parameters are used for the various failure modes (e.g., failure to run and failure to start) of various components.

c. Provide the basis for the omission of the following components from the CCF analysis or for their inclusion into "the generic component:" circuit breakers (AC, DC, reactor trip breakers excluded); relays (engineered safety features actuation system (ESFAS)); and electrical switchgear.

d. Please discuss the impact of the above assumptions on the CDF, dominant accident sequences, and your conclusions regarding plant vulnerabilities.

e. Provide a discussion of how the common cause losses of AC and DC buses were treated as initiating events.

5. Please discuss why the main feedwater (MFW) system, or parts thereof (e.g., condensate and feedwater booster pumps) cannot be credited in a small loss of coolant accident (LOCA). Are there procedures to shut off the MFW after a small LOCA and how is this modeled in the IPE? What are the associated human error probabilities (HEPs)? Why can't the MFW (or the associated condensate and feedwater booster pumps) be used in the same manner as the EFW in small LOCAs?

6. Describe the process used to identify and account for Interfacing System LOCAs (ISLOCA). Discuss the most likely flow paths and the impact of the degradation or loss of mitigating systems due to ISLOCA.

7. This question concerns your treatment of flooding.

   The submittal indicates that all but one flood zone were eliminated from further consideration through a qualitative analysis. The submittal describes the qualitative screening process, but it is not clear if certain assumptions were used for eliminating some zones, e.g., the three areas where "either no water sources were found or the sources were out of range of safe shutdown equipment and reactor trip components," or some rooms where no reactor trip is said to occur (e.g., the 480V switchgear room, the heating ventilation and air conditioning (HVAC) chilled water pump area, the local relay panel room, etc.). For the one area included in the analysis, the intermediate building elevation 412' general floor area, room 12-02, it is not clear if assumptions were made with respect to flow rates and flood isolation timing, effect of spray/flooding on other parts of the component cooling water (CCW) or service water (SW) system and spray/flooding effects on other safety components in the area.

The internal flooding CDF is 1.5E-6. The flooding is caused by ruptures in the SW or CCW systems in the intermediate building elevation 412' general floor area, room 12-02. Ruptures in the CCW system are calculated to occur with a frequency of 9.9E-4/yr, giving a CDF of 1.2E-6/yr, resulting in a conditional CDF (CCDP) of 1.2E-3.

For the flooding initiator loss of train A of SW (caused by SW ruptures), the initiating event frequency is 1.1E-4/yr, the CDF is 2.9E-7/yr and the CCDP is 2.6E-3; for loss of train B of SW, the IE frequency is 1.0E-4/yr, the CDF is 1.9E-8/yr and the CCDP is 1.9E-4. It should be noted that, according to the submittal, these events result in a loss of a single train of CCW or SW, with no other failures. However, if it is possible to disable the whole SW system, with no additional failures, the flooding CDF from that scenario would potentially rise, as the CCDP from the internal events analysis for total loss of SW is 0.48 (initiating event frequency of 3.6E-5/yr and CDF of 1.74E-5/yr), i.e., 2-3 orders of magnitude higher than in the flooding analysis.

Additionally, the CCDP and the CDF from all flooding scenarios might rise if additional safety equipment in the room can be affected. A statement is made that no other equipment (except for the leaking component) would be affected in the flooding scenario, because the flood water would flow into the tendon jacking area and the tendon access gallery. It should be noted that there are several safety components in the room (battery charger, CCW pumps, all motor-driven and steam-driven EFW pumps, SW booster pumps, CCW cross-connection valves). None of these components is spray protected. It is not clear if there are assumptions in the IPE regarding isolation of flooding in a certain time period to prevent these components from being flooded (i.e., assumptions regarding the flow rate vs. the capacity of the tendon jacking area and the tendon access gallery). There may also be additional assumptions made in the IPE regarding the spray effect.

a. Please describe the process addressing the zones with safety-related equipment considered, the types of flood initiators in these zones, and the frequency of the initiators. Provide enough details of your flooding analysis to understand the scenarios more fully, e.g., equipment in the room, equipment affected by the scenario, probability of total loss of SW or CCW in the one zone admitted into the flooding analysis, response of the plant, existence of any flood barriers, quantification of various components of your flood scenario, assumptions made. Discuss your consideration of drains (including backflooding to other areas and probability of failure, i.e., due to blockage), separation, doors allowing flood propagation to other areas, credit given for actions by operators to stop the flood or to mitigate the consequences and the specific criteria used to eliminate each zone.

b. Please describe your treatment of the spray effect resulting from the spurious actuation of the fire suppression equipment in your flood scenarios.

c. Discuss how maintenance errors were treated in the flooding analysis. Include errors committed while in cold shutdown, which were left undiagnosed until the flood event occurred while the unit is at power.

8. It is not apparent that loss of HVAC has been considered, either as an initiator or as part of subsequent failures. Given the function of this system and given the fact that this system has been found to be important to risk at some plants, please provide a more detailed discussion of your investigation into the impact of loss of HVAC in rooms containing safety-related equipment, including rooms with pumps, electrical equipment, and the control room.

   a. Provide a discussion of loss of HVAC both as an initiating event and as a failure subsequent to an initiator. Your discussion should include the following: systems in the areas considered; basis for elimination, describing the method of assessment, including calculations; credited operator actions; alarms; procedures; and staged equipment.

   b. Please consider the fact that upon loss of room cooling equipment may be isolated prior to reaching the damaging temperature. Also consider that, if the damaging temperature is reached, timely recovery of such equipment should probably not be credited.

   Provide the impact of your consideration on the results (CDF, important sequences) and the Fussell-Vesely importance of the HVAC failures (including maintenance) to the total CDF.

9. The transient initiating event frequency reported in the submittal seems high, especially for certain categories (e.g., spurious safety injection signal 0.57/yr, positive reactivity insertion 0.56/yr, total or partial loss of feedwater flow 2.77/yr, etc.). It should also be noted that the on-line maintenance unavailability of the chilled water system, which contributes to the initiating event frequency of 1.8E-2/yr, is relatively high. In view of the fact that transients contribute 40% to the reported CDF of 2.E-4/yr, are there any systematic programs in place to reduce the transient frequency, and if so, how effective are they? Please discuss the plant-specific data used for calculation of these initiating event frequencies and share any insights as to why the frequency numbers are so high.

10. The calculated CDF from internal events is relatively high (2.E-4/yr). The submittal treats the issue of vulnerabilities by subdividing the CDF into several groups, subdividing the groups into subgroups, applying Nuclear Management & Resources Council (NUMARC) criteria to these subgroups and occasionally qualitatively taking credit for unscheduled future improvements to bring the subgroup CDF below NUMARC

guidelines ("10^{-5} to 10^{-6} range"). For instance, group IIA (induced LOCA with loss of primary coolant makeup or adequate heat removal in the injection phase) contributes 64% to the CDF (or 1.3E-4/yr). It is further subdivided into several subgroups, one of which, SBO, contributes 21% to the total CDF. Here, credit is taken for a future (date unspecified) installation of the new reactor coolant pump (RCP) seal O-rings and a future (date unspecified) consideration of fire service system connection to RCP seal thermal barrier cooling.

Another way to look at vulnerabilities would be to scan the table of top event (Fussell-Vesely) importance, which shows the contribution of various failures to the CDF. The chilled water system is the top contributor, at 39% importance, followed by diesel generators at 39%, followed by SW at 27%. Chilled water is important because it provides cooling for CCW and charging pumps, as well as HVAC for important safeguards systems throughout the plant. SW is a support for both component cooling and chilled water systems. The plant has another dependency that most other plants do not, i.e., RCP seal thermal barrier cooling depends on offsite power. Some of these dependencies may have been addressed by recent post-submittal improvements (see related question).

Also, it could be noted that the IPE submittal's results point out a high or relatively high conditional core damage probability (CCDP) for certain initiators, e.g., a CCDP of 0.48 for loss of SW, 3.7E-2 for loss of two 120V AC panels, 1.1E-3 for loss of offsite power.

In view of the above, please justify your treatment of the vulnerability issue. Please consider the situation where the CDF groups are not divided into subgroups or where a definition of vulnerabilities similar to the one mentioned above were used, and state whether or not vulnerabilities (and which ones) would exist for your plant. Please also state what would be the actions taken to address such vulnerabilities and what would be the impact of these actions on your results.

11. NUREG-1335, Section 2.1.6 part 4 requests "a thorough discussion of the evaluation of the decay heat removal function." Section 3.4.3, Decay Heat Removal (DHR) Evaluation, deals with this issue. However, certain details are missing. Please provide the contribution of DHR and its constituent systems (including feed and bleed) to CDF and the relative impact of loss of support systems on the frontline systems that perform the DHR function.

12. It is not always clear from the IPE submittal whether the plant improvements described are being proposed for further consideration or were actually implemented. For example, in Section 3.4.2, Vulnerability Screening, mention is made of the new RCP seal O-rings and a fire water connection for RCP thermal barrier cooling, yet these actions do not appear on the list of improvements (proposed and/or implemented) in Section 6.1, Plant Improvements. In addition, please provide the details of certain recent improvements which may address

the chilled water dependency of the charging and CCW systems (apparently, chilled water is no longer used for CCW motor cooling, i.e., CCW now cools itself, and charging pumps are no longer cooled by chilled water but by CCW).

Please provide the following, if not already provided, regarding each improvement:

a. The specific improvements that have been implemented, are being planned or are under evaluation.

b. The status of each improvement, i.e. whether the improvement has actually been implemented already, is planned (with scheduled implementation date), or is under evaluation.

c. The improvements that were credited in the reported CDF.

d. If available, the reduction to the CDF or the conditional containment failure probability that would be realized from each plant improvement if the improvement were to be credited in the reported CDF (or containment failure probability), or the increase in the CDF or the conditional containment failure probability if the credited improvement were to be removed from the reported CDF (or containment failure probability).

e. The basis for each improvement, i.e. whether it addressed a vulnerability, was otherwise identified from the IPE review, was developed as part of other NRC rulemaking (such as the SBO Rule), etc.

13. In your discussion of core cooling success criteria, it is stated that the operators are not required to initiate the inadequate core cooling procedure until the steam temperature in the vessel is greater than 700°F (in conjunction with a low reactor vessel level indication system (RVLIS) indication). However, at this temperature, the pressure is 3100 psi, and at 705°F, the critical temperature for saturated steam, the pressure is over 3200 psi. It is not clear whether the steam is saturated or superheated. Please clarify. If the steam is saturated, please verify that the limiting reactor coolant system (RCS) pressure (e.g., used in your anticipated transient without scram (ATWS) analysis) is above 3200 psi, and discuss what bearing, if any, this pressure has on the procedure addressing inadequate core cooling. Also please verify that the correct core cooling success criterion used in your Modular Accident Analysis Program (MAAP) analyses is 1200°F for maximum fuel cladding temperature, not for core exit temperature as stated in the submittal.

14. The success criteria used in the Summer IPE submittal are more optimistic than the ones traditionally used for certain accidents in previous probabilistic risk assessments (PRAs). The success criteria in the submittal can be compared to those in the NUREG/CR-4550 analyses, specifically the ones for Surry, as the two plants have

certain outward similarities (both are 3-loop Westinghouse PWRs of similar power ratings, with Summer having a higher rating, and the containments are of equal size though Surry's is subatmospheric). In addition the LOCA size ranges for LOCA categories are the same as in the NUREG/CR-4550 analysis for Surry (with the minor variation that your small LOCAs also encompass the very small LOCAs defined for Surry, i.e., RCP seal failures, etc.). The specific instances where your success criteria differ are noted below:

- for large LOCAs, short term success is defined as either one residual heat removal (RHR) pump (with no accumulators needed) or one high head injection (HPI) pump in conjunction with 2/2 accumulators. No early containment pressure suppression is required. The success criterion used in the NUREG/CR-4550 requires 2/2 accumulators in addition to the RHR pump, and the HPI pump is not be credited. Also, early containment pressure suppression is required for containment integrity.

- for medium LOCAs, the submittal's success criteria indicate that the RCS pressure will stay above the RHR pump shutoff head, unless action is taken to depressurize via the secondary side. In the NUREG/CR-4550 analysis, medium LOCA will lead to a quick RCS depressurization, such that one RH pump <u>and</u> two accumulators are needed, in addition to the one HPI pump. Early containment pressure suppression is needed, as well.

a. Are these novel success paths reflected in the emergency procedures, and if not how were the HEPs estimated for actions needed in these success paths?

b. What is the impact on timing of operator actions if containment sprays come on in certain LOCAs? Are the operators instructed to shut off the sprays to conserve the refueling water storage tank (RWST) water? If yes, how are such actions accounted for in the model? If not, does the timing of your scenarios account for the fact that the RWST will run out of RWST sooner than anticipated because both trains of sprays might come on automatically?

c. If available, please estimate the impact of these novel success paths on your results (CDF, important sequences).

15. Please provide the following information missing from the submittal:

a. What are the success criteria for the running time of the diesel generators in a loss of offsite power accident and what is the basis?

b. What is the assumed failure pressure of the RCS in an ATWS and what is the basis?

B. Human Reliability Analysis (HRA)

1. It is not clear from the submittal whether the risk impact of the human potential to cause an accident was considered. Identification of the pre-initiator human events that can disable a system, such as failure to properly restore after maintenance or miscalibration of instrumentation, are essential to the HRA. Section 3.3.3, "Human Failure Data," of the submittal and associated Table 3.3.3-2 "Operator Actions Important to System Operation" (including system alignment and system actuation actions), Table 3.3.3-3 "Operator Actions Important to Restoration of Failed Equipment," and Table 3.3.3-5 "Operator Actions Related to System Alignment" do not show the relative risk associated with pre-initiator human actions. Please provide a list of the types of pre-initiator human events, in order of decreasing importance, that were considered in the analysis.

2. If the submittal does include pre-initiators human actions, it is important to describe the process used to identify and select the important pre-initiators involving miscalibration of instrumentation and the failure to properly restore to service after test or maintenance. The process used to identify and select the instrumentation calibration related human action events may include the review of procedures, and discussions with appropriate plant personnel on interpretation and implementation of the plant's calibration procedures. For assessing the failure to restore important equipment to service after test or maintenance, the process may include the review of maintenance and test procedures, and discussions with appropriate plant personnel on the interpretation and implementation of the plant's test and maintenance procedures. In Section 3.3.3, "Human Failure Data," of the submittal, there is no description of the process used to identify and select the pre-initiators human actions in Tables 3.3.3-2 "Operator Actions Important to System Operation" (including system alignment and system actuation actions), and 3.3.3-5 "Operator Actions Related to System Alignment." Please provide a description of the process that was used to identify pre-initiator human actions involving miscalibration of instrumentation and failure to restore equipment to service after test or maintenance. In addition, please provide examples illustrating the processes using several relatively important pre-initiator human actions.

3. It is not clear from the submittal what screening values were used and the bases for the values. In Section 3.3.3, "Human Failure Data," there is no description of any screening values or process used to identify and select the pre-initiators human actions. Please provide all of the screening value(s) used and the basis for the value(s); i.e., provide the rationale of how the selected screening value(s) did not eliminate (or truncate) important pre-initiator human events. In addition, please provide a list of actions initially considered and those screened.

4. The submittal does not clearly identify the actual recovery factors applied in quantifying the pre-initiator human events. Factors that are used to modify the generic basic human error probability (BHEP) can include, for example, post-maintenance or post-calibration tests, daily written checks, independent written verification checks, administrative controls, etc. In Section 3.3.3, "Human Failure Data," of the submittal, there were no pre-initiator recovery factors mentioned. If they are used, please provide a list of recovery factors considered, their associated values, and provide specific examples illustrating their use. Also, if used, please provide a concise discussion of the justification and process that was used to determine the appropriateness of the recovery factors utilized.

5. It is not clear from the submittal how dependencies associated with pre-initiator human errors were addressed and treated. There are several ways dependencies can be treated. In the first example, the probability of the subsequent human events is influenced by the probability of the first event. For example, in the restoration of several valves, a bolt is require to be "tightened." It is judged that if the operator fails to "tighten" the bolt on the first valve, he will subsequently fail on the remaining valves. In this example, subsequent HEPs in the model (i.e., representing the second valve) will be adjusted to reflect this dependence. In the second example, poor lighting can result in increasing the likelihood of unrelated human events; that is, the poor lighting condition can affect different operators' abilities to properly calibrate or to properly restore a component to service, although these events are governed by different procedures and performed by different personnel. This type of dependency is typically incorporated in the HRA model by "grouping" the components so they fail simultaneously. In the third example, pressure sensor "x" and "y" may be calibrated using different procedures. However, if the procedures are poorly written such that miscalibration is likely on both sensor "x" and "y", then each individual HEP in the model representing calibration of the pressure sensors can be adjusted individually to reflect the quality of the procedures. The submittal in Section 3.3.3 contains a two- paragraph discussion on "Dependent Events" without any reference to pre-initiator human actions. Tables 3.3.3-2 "Operator Actions Important to System Operation" (including system alignment and system actuation actions), and 3.3.3-5 "Operator Actions Related to System Alignment" do not indicate anything about the implementation of pre-initiator human action dependencies. Please provide a concise discussion of how dependencies were addressed and treated in the pre-initiator HRA such that important accident sequences were not eliminated. If dependencies were not addressed, please justify.

6. The submittal is not clear about the risk significance of human actions to contribute to, and mitigate the consequences of an accident. Table 3.3.3-4, "Human Reliability Quantification Results" of the submittal does not contain this important information. Table 3.4.1-3, "Human Error Probability (HEP) Sensitivity Analysis Dominant Sequences" provides the post-initiators increased by an order

of magnitude to maintain the relative importance of these actions. Table 1.5.1-3, "Key Contributors to Dominant Accident Sequences" of the submittal is not detailed enough to provide this information. Please provide a list of the most important risk significant post-initiator human actions and their associated HEPs in the most important sequences in which they appear.

7. The submittal does not clearly describe the type of human errors considered for each post-initiator human event identified. For example, a human event identified may be the failure to feed and bleed, while the types of human errors considered may involve failure to open the correct valve (error of omission), or opening an incorrect valve (error of commission). No mention of types of human errors was found in the submittal's Section 3.3.3, "Human Failure Data." Please identify what types of human errors were considered for the human event identified.

8. The submittal does not clearly describe the method used to identify and select response type actions and recovery type actions for analysis. The method utilized should confirm the plant emergency procedures, design, operations, and maintenance and surveillance procedures were examined and understood to identify potential severe accident sequences. The submittal's Section 3.3.3, "Human Failure Data" and associated Table 3.3.3-4, "Human Reliability Quantification Results" are not clear on the identity of the response type actions and recovery type actions used. Also, the method used was not addressed. Please provide a description of the process that was used for identifying and selecting the response and recovery type actions evaluated.

9. The submittal does not clearly indicate whether a screening process was utilized to help differentiate the more important post-initiator human events. No mention of screening post-initiator human errors was found in the submittal's Section 3.3.3, "Human Failure Data." If a screening process was used, please provide all of the screening value(s) used and the basis for the value(s); i.e., provide the rationale for how the selected screening value did not eliminate (or truncate) important human events. Also, provide a list of errors initially considered and those screened. If a screening process was not used, please identify the more important post-initiator human events.

10. In applying performance shaping factors (PSFs), the consideration of time is important. The submittal is not clear on how available time and "required" time were calculated for the various post-initiator human events. "Required" time is the time needed for an operator to diagnose and perform the actions. Table 3.3.3-4, "Human Reliability Quantification Results," of the submittal is a summary of all the HEPs used to support the VCSNS accident sequence quantification. This includes the results of the technique for human error rate prediction (THERP) analysis and conditional analysis. The "Time Window" specified in the table is the time from initial indication that action

is required until the operator action must be completed for success of the action. For several of the important post-initiator human events examined, provide the available and "required" times estimated for the operator action and the bases (e.g., calculated from simulator exercises, estimated from walkdowns) for the time chosen. Also provide illustrations of how different times were calculated for the same task but in different sequences.

11. It is not clear from the submittal what plant-specific PSFs were used to modify the BHEP and what the bases were for reducing HEPs through their application. The plant-specific information could include the size of crew, availability of procedures, time available, time required, etc. The process could include an examination of procedures, training, human engineering, staffing, communication, and administrative controls.

    The submittal in Section 3.3.3 briefly states that "... PSFs were also used concurrently to modify the nominal HEP (that is, the probability of a given human error when the effects of plant-specific PSFs have not yet been considered)." and is not mentioned again. Please provide a list of the types of plant-specific PSFs considered and their values, and discuss by way of example how these PSFs were used to modify the BHEPs of important post-initiator human events.

12. The submittal is not clear whether response type actions and recovery type actions were considered. Response type actions include human actions performed in response to the first level directive of the emergency operating procedures (EOPs). For example, suppose the EOP directive instructs the operator to determine reactor water level status, and another directive instructs the operator to maintain reactor water level with system X. These actions - reading instrumentation to determine level and actuating system X to maintain level - are response type actions. Recovery type actions include those performed to recover a specific failure or fault and may not be "proceduralized." For example, suppose the EOP directive instructs the operator to maintain level using system x, but the system fails to function and the operator then attempts to recover it. This action - diagnosing the failure and then deciding on a course of action to "recover" the failed system - is a recovery type action. Section 3.3.3 has not made a distinction between the response type actions and recovery type actions used as defined above. Please provide separate lists of the response and recovery actions considered in the analysis. If response or recovery actions were not considered, please justify. Also justify why recovery actions, if used, are not proceduralized.

13. It is not clear from the submittal how dependencies were addressed and treated in the post-initiator HRA. The performance of the operator is both dependent on the accident under progression and the past performance of the operator during the accident of concern. Improper treatment of these dependencies can result in the elimination of potentially dominant accident sequences and, therefore, the identification of significant events. The submittal in Section 3.3.3

contains a two paragraph discussion on "Dependent Events" without any reference to post-initiator human actions. Tables 3.3.3.1, "Operator Actions Evaluated in Plant Response Trees and their Initiating Event Contexts," 3.3.3.3, "Operator Actions Important in Restoration of Failed Equipment" and 3.3.3.4, "Human Reliability Quantification Results" do not indicate anything about the implementation of post-initiator human action dependencies. Please provide a concise discussion and examples illustrating how dependencies were addressed and treated in the post-initiator HRA such that important accident sequences were not eliminated. If the submittal did not address dependencies in the quantification, please justify. The discussion should address the two points below:

a. Human events are modeled in the fault trees as basic events such as failure to manually actuate. The probability of the operator to perform this function is dependent on the accident in progression - what symptoms are occurring, what other activities are being performed (successfully and unsuccessfully), etc. When the sequences are quantified, this basic event can appear, not only in different sequences, but in different combinations with different systems failures. In addition, the basic event can potentially be multiplied by other human events when the sequences are quantified which should be evaluated for dependencies.

b. Human events are modeled in the event trees as top events. The probability of the operator to perform this function is still dependent on the accident progression. The quantification of the human events need to consider the different sequences and the other human events.

14. The submittal in Section 3.3.3 noted that "operator actions important to equipment restoration were assessed via expert judgment." Please describe the expert judgment process and who the experts were to render such judgment by stating their number and individual qualifications.

C. <u>Back-End Analysis</u>

1. It is stated in the IPE submittal that "Comparison of VCSNS and Zion cavity/instrument tunnel designs clearly indicates that the VCSNS geometry would trap and de-entrain more debris than in the Zion configuration." Please provide a more detailed discussion by comparing the cavity configurations of the two plants and pointing out the similarities and differences that support the above claim.

2. According to the IPE, the majority of the core debris will be de-entrained by containment structures and remain in the cavity during high pressure melt ejection (HPME). Please provide a more detailed discussion of the debris flow path and debris distribution during HPME. Please include in the discussion the effect of the two cavity cooling fan openings, which, according to the IPE submittal, connect the lower compartment and the reactor cavity (through the instrument

tunnel, Figure 4.2.1-5) and allow water on the lower compartment floor to flow to the reactor cavity. Please discuss the effect of these openings on the dispersion and distribution of core debris after vessel melt-through and, consequently, any adverse effect on containment integrity or equipment availability.

3. In the IPE, molten core-concrete interaction (MCCI) was evaluated using a simple bounding analysis model to determine whether the aggressive attack on concrete by molten core debris could lead to late containment failure. The conclusion in the IPE is that molten core-concrete can be excluded from consideration as a significant late containment failure mechanism. Keeping in mind the assumed maximum coolable debris depth of 25 cm mentioned in Generic Letter 88-20, please discuss the depth of core debris in the VCSNS cavity and the effect of non-uniform spread of debris on debris coolability.

4. The second paragraph on page 4-26 of the submittal states that "in the absence of external cooling of the RPV, relocation of the molten core debris into the lower head is assumed to lead directly to failure of the reactor vessel; no attempt is made to take credit for potential in-vessel recovery." However, in the VCSNS water can flow to the reactor cavity from the containment floor and the lower part of the vessel can be submerged. Please discuss the likelihood of a submerged vessel for VCSNS and the effect of omitting this external cooling when defining the source term. Since this mechanism may delay, if not terminate, vessel penetration, fission product production and release paths are affected (e.g., in-vessel release from a dry debris bed versus ex-vessel release from a debris bed covered by water). The release of fission products to the environment may actually increase if the containment fails and external cooling was accounted for in the source term calculation. Please also discuss the effect of external vessel cooling (which results in maintaining the RCS at high temperature for a longer time) on the probability of creep rupture of RCS boundaries and steam generator tubes, and consequently, the effect on containment performance and source terms for VCSNS.

5. Containment isolation status is one of the VCSNS Plant Response Tree top events. It is also indicated by the sixth digit in the 7-digit plant damage state (PDS) designator. With respect to the analysis of containment isolation failure probability, NUREG-1335 (Section 2.2.2.5, page 2-11) states that "the analyses should address the five areas identified in the Generic Letter, i.e., (1) the pathways that could significantly contribute to containment isolation failure, (2) the signals required to automatically isolate the penetrations, (3) the potential for generating the signals for all initiating events, (4) the examination of the testing and maintenance procedures, and (5) the quantification of each containment isolation failure mode (including common-mode failure)." Please discuss your findings related to all of the above five areas.

6. Temperature-induced steam generator creep rupture, which is considered in other IPEs, is not addressed in the VCSNS IPE. In some IPEs, the

probability of induced steam generator tube rupture (SGTR) increases as the RCP is restarted following the direction of procedures. Please discuss the probability of induced SGTR. Please include in the discussion the probability of RCP operation and the effect of RCP operation on the probability of induced SGTR.

7. In NUREG-1335 it is stated that "documentation should be provided to support the availability and survivability of systems and components with potentially significant impact on the containment event tree (CET) or the radionuclide release." This issue is not discussed in the back-end part of the VCSNS submittal. Please discuss the survivability of the equipment under severe accident conditions. Please include in the discussion the environmental conditions (e.g., temperature, pressure, radiation, aerosol plugging, and debris effects) derived and used in the evaluation.

8. The Generic Letter CPI recommendation for PWR dry containments is the evaluation of containment and equipment vulnerabilities to localized hydrogen combustion and the need for improvements (including accident management procedures).

   Please discuss whether plant walkdowns have been performed to determine the probable locations of hydrogen releases into the containment. Discuss the process used to assure that : (1) local deflagrations would not translate to detonations given an unfavorable nearby geometry, and (2) the containment boundary, including penetrations, would not be challenged by hydrogen burns.

   Please identity potential reactor hydrogen release points and vent paths. Estimates of compartment free volumes and vent path flow areas should also be provided. Please specifically address how this information is used in your assessment of hydrogen pocketing and detonation. Your discussion (including important assumptions) should cover likelihood of local detonation and potentials for missile generation as a result of local detonation.

9. In the Summer IPE ten source term bins (STBs) are selected for source term calculations. Among these ten STBs, three involve containment bypass or isolation failure. Of the remaining seven STBs, only one involves containment failure (STB 5, with a frequency of 4.1E-5, contributing 20% of total CDF, Tables 4.4.3-2 and 4.4.4-3). However, according to the results of the source term calculation, the release fraction of volatile fission products for STB 5 is the lowest among all STBs (Tables 4.4.4-1 and 4.4.4-3). The release fraction for STB 5 is significantly less than that of some STBs in which the containment remains intact. It seems that the operation of containment spray is one of the primary reasons for the small release. Please discuss the dependence of the spray system on other support systems (e.g., cooling requirement) and the availability of the spray system (while emergency core cooling system (ECCS) is not available) for the sequences binned in STB 5. Please discuss the models used for the sprays for fission product scrubbing, the uncertainty of these

models and their effects on the uncertainty of fission product releases. Please also discuss the long-term releases (beyond the calculation time of 48 hours) and the effect of revaporization of fission products on long-term releases. Since containment failure area is also an important factor, please estimate the change in source terms by the use a containment failure area greater than that used in the IPE to address the uncertainty in containment failure area (e.g., a failure area of the order of 1 ft^2. The IPE uses 0.03 ft^2 for the base case and 0.1 ft^2 for the sensitivity case.).

10. A containment failure area of 0.03 ft^2 is assumed in the VCSNS IPE. This seems to be based on the leak-before-break behavior assumed to occur at slow pressurization. In the VCSNS IPE, containment integrity is assumed to be maintained before containment pressure reaches a containment failure pressure of 142 psig. Since it is more likely to have a larger containment failure area at higher containment pressure, please discuss the dependence of containment failure area on containment pressure load for VCSNS. Please address uncertainties in the discussion.

11. There are a number of items in the level 2 part of the submittal which need clarification:

    a. The last paragraph of Section 4.4.4 (page 4-55) states that "lastly, Table 4.4.4-4 summarizes the source-term results by release category and shows the conditional probability of each release category given core damage. These results show that should a core damage event occur at VCSNS, there is more than a 97-percent probability that the radionuclide release would represent less than or equal to 0.01 percent of the volatile fission products." The values quoted here do not seem to be consistent with the numbers presented in Table 4.4.4-4. Please clarify.

    b. Sequence 98 is initiated by a total loss of SW with containment isolation failure. Results of MAAP calculation for this case are presented in both Table 4.4.4-1 (for the base case) and 4.5.2-1 (for the base case in the sensitivity analyses). Although results presented in these two tables are in general similar, they are not the same. Results presented in Table 4.5.2-1 show the occurrence of hydrogen burns and a much higher containment temperature, while Table 4.4.4-1 does not show hydrogen burns and a lower containment temperature. Please explain the discrepancy.

    c. It seems that the "Analyzed Functional Sequence" for Source-Term Bin 5 should be TRE13IH instead of TRE12IH. Please clarify.

Mr. Gary J. Taylor
South Carolina Electric & Gas Company

VIRGIL C. SUMMER NUCLEAR STATION

cc:

Mr. R. J. White
Nuclear Coordinator
S.C. Public Service Authority
c/o Virgil C. Summer Nuclear Station
Post Office Box 88, Mail Code 802
Jenkinsville, South Carolina 29065

J. B. Knotts, Jr., Esquire
Winston & Strawn Law Firm
1400 L Street, N.W.
Washington, D.C. 20005-3502

Resident Inspector/Summer NPS
c/o U.S. Nuclear Regulatory Commission
Route 1, Box 64
Jenkinsville, South Carolina 29065

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., N.W., Ste. 2900
Atlanta, Georgia 30323

Chairman, Fairfield County Council
Drawer 60
Winnsboro, South Carolina 29180

Mr. Virgil R. Autry
Director of Radioactive Waste Management
Bureau of Solid & Hazardous Waste Management
Department of Health & Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Mr. R. M. Fowlkes, Manager
Nuclear Licensing & Operating Experience
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065