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MEMORANDUM FOR: Elinor Adensam, Chief, Licensing Branch #4
~~Division of Licensing~~

FROM: Brian W. Sheron, Chief, Reactor Systems Branch
Division of Systems Integration

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - VOGTLE ELECTRIC GENERATING STATION

Plant Name: Vogtle Electric Generating Station, Units 1 and 2
Docket No.: 50-424/425
Licensing Status: OL
Responsible Branch: Licensing Branch #4
Project Manager: M. Miller
Review Status: Request for Additional Information

Enclosed with this letter is a set of questions concerning the Vogtle plant. These questions are a result of a review of those sections of 5.2 and 5.4 of the FSAR for which Reactor Systems Branch has primary review responsibility. RSB is continuing its review and will submit additional questions as the evaluation proceeds through the other areas for which we are responsible.

Original signed by:
Brian W. Sheron

Brian W. Sheron, Chief
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Enclosure:
As stated

cc: R. W. Houston
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GEORGIA POWER COMPANY

VOGTLE ELECTRIC GENERATOR PLANT, UNITS 1 AND 2

DOCKET NOS. 424/425

REQUEST FOR ADDITIONAL INFORMATION

1. An examination of P&ID's shows, in general, that there are no coordinates on the figures; consequently, it is very difficult to locate equipment and interconnections from figure to figure. This is not acceptable; figures must be corrected so that coordinates are available. Correct all piping and instrument drawings (P&ID's), accordingly.

2. (5.2.2.1) What events, other than those listed could lead to overpressurization of the RCS if adequate overpressure protection were not provided?

3. (5.2.2.8, 5.2.2.10) What kinds of positive position indication are provided for the pressurizer safety valves and PORVs? Discuss compliance with NUREG-0737 items II.D.1 and II.D.3.

4. (5.2.2.10.2) What are the postulated worst case mass input and heat input events for a Low Temperature/Overpressurization event? Staff position, for previous Westinghouse plants, has been that the

design basis mass addition event is the inadvertent actuation of a safety injection pump under runout conditions. Justify your selection if other than the above.

5. (5.2.2.10.4) A number of administrative controls have been described in the FSAR to maintain RCS pressure to within allowable limits. What alarms are available in the control room to remind the operator that specific administrative controls are to be effected (e.g., maintaining at least one RCP in operation until reactor coolant temperature reaches 160°F and maintaining a full open valve in the bypass line to the letdown orifices during water solid operation)?

6. (5.2.2.10.4) In the FSAR it is noted that ECCS actuation on high pressurizer pressure or low steam line pressure is blocked for RCS pressures under 1900 psig. Provide an analysis that shows that these SI signals are not needed for this condition.

7. (5.2.2) Section 5.4.13.2 describes the loop seals on the pressurizer safety valves. Has the delay due to the time it takes to discharge the water from these loop seals been accounted for in the limiting pressure transient? If it has not been accounted for, how would this delay affect the conservatism of the result?

8. (5.2.2, 5.4.12, 6.3) Check valves in the discharge side of the high pressure safety injection, low pressure safety injection, RHR, and charging systems perform an isolation function in that they protect low pressure systems from full reactor pressure. The staff requires that these check valves be classified ASME IWV-2000 category AC, with the leak testing for this class of valve being performed to code specifications. It should be noted that a testing program which simply draws a suction on the low pressure side of the outermost check valves will not be acceptable. This only verifies that one of the series check valves is fulfilling an isolation function. The necessary frequency will be that specified in the ASME Code, except in cases where only one or two check valves separate high to low pressure systems. In these cases, leak testing will be performed at each refueling after the valves have been exercised. Identify all check valves which should be classified Category AC as per the position discussed above. Verify that you will meet the required leak testing schedule, and that you have the necessary test lines to leak test each valve. Provide the leak detection criteria that will be in the Technical Specifications.

9. (5.2.2) WCAP 7769, Section 3.4 assumes failure of one steam generator safety relief valve per loop. Provide assurance that your remaining safety valves can provide the required minimum capacity or justify why your analysis assumes only a single failure in one loop.

10. (5.2.2.2) It is stated in 5.2.2.2 of the FSAR that the pressurizer and SG safety valves are sized with sufficient capacity to provide overpressure protection for typical worst-case transient conditions. What are these conditions? Verify that sufficient margin has been provided to account for uncertainties in the design and operation or the plant and that the maximum instrumentation and control errors have been assumed. Discuss the preoperational tests which will verify the accuracy of instrumentation systems used to initiate overpressure protection.

11. (5.2.2) What are the setpoint tolerances for all of the safety and power operated valves? What tolerances are taken credit for in the setpoint analyses? Do the analyses take into account setpoint drift?

12. (5.2.2.2) In Section 5.2.2.2, reference is made to WCAP-7769, Revision 1. Provide a comparison of Vogtle parameters with all parameters listed in Table 2.2 of this topical report. Where differences exist, show that these differences will not affect the conservatism of the results given in WCAP-7769.

13. (5.2.2) Provide verification that your analysis of the limiting transient for overpressure protection assumes the reactor trip is initiated by the second safety grade signal.

14. (5.2.2) Provide assurance that the dynamic loading of the PORVS and the code safeties due to water relief has been considered in the piping and support analysis including the passage of a water slug and effects of water hammer. What liquid water relief rates were assumed in the loading analysis? Are these values consistent with experimental results? Are the power operated relief valves and safety valves designed and qualified for liquid relief?

15. (5.2.2, 5.4.13) Section 5.4.13.2 cites a backpressure compensation feature on the pressurizer safety valves. Provide a discussion of this feature which explains how this function is performed.

16. (5.2.2) Have the pressurizer PORVs been qualified for the dynamic loads that could be sustained when the maximum liquid flow rate or maximum acceleration of liquid occurs during a low temperature overpressurization?

17. (5.2.2.10) In Section 5.2.2.10, it is stated that the low temperature overpressure protection system is manually armed. Is there an alarm to alert the operator to arm the system at the correct plant condition during cooldown as required by Branch Technical Position RSB 5-2.B.3?

18. (5.2.2) Provide a description of the design features to be used to mitigate the consequences of overpressure events while

operating at low temperatures. Our position regarding overpressure protection while operating at low temperatures is presented in the Branch Technical Position RSB 5-2 attached to SRP 5.2.2. Your description should address each portion of this position.

19. (5.2.2) The Branch Technical Position RSB 5-2 states the reactor vessel overpressurization protection system should meet the single active failure criterion when the initiating cause of the event is not considered as the single active failure. Provide a failure modes and effects analysis to demonstrate that a single electrical or mechanical component failure will not disable both trains of PORVs from functioning.

20. (5.2.2.10.1) In Section 5.2.2.10.1 of the FSAR, you indicate that "an auctioneered system temperature is continuously converted to an allowable pressure and then compared to the actual RCS pressure. The system logic first annunciates a main control board alarm whenever the measured pressure approaches within a predetermined amount of the allowable pressure, thereby indicating that a pressure transient is occurring, and on a further increase in measured pressure, an actuation signal is transmitted to the PORVs when required to mitigate the pressure transients." Our review of the low temperature overpressure protection design for certain other Westinghouse plants indicates that a failure in

the temperature auctioneer for one PORV (signalling it to remain closed) could also fail the other PORV closed (by denying its permissive to open). Address this concern about a potential common mode failure in the low temperature overpressure protection system for Vogtle.

21. (5.2.2) Provide your limiting Appendix G curve for the first eighteen full power months of operation. Discuss the operational procedures which will minimize the likelihood of an overpressure event.

22. (5.2.2.10.1) The staff is concerned that your proposed low temperature overpressure protectio- (LTOP) system does not adequately protect the reactor vessel during transient events where the vessel wall temperature lags behind the temperature used in the variable setpoint calculator. For example, starting an RCP in a loop with a hot steam generator when the RCS is water solid causes the RCS pressure and temperature to rise. Your LTOP system would automatically raise the PORV setpoint as a function of auctioneered cold or hot leg temperature, but the vessel wall will not be heated in this transient at the same rate. Thus, due to the LTOP system auctioneering scheme, the part of the RCS most vulnerable to brittle fracture will be protected to a higher pressure than its temperature allows.

If, during a cooldown, the cold leg temperature detector downstream of the generator(s) being used failed, and a mass input event occurred, your proposed LTOP system may not protect the coldest location in the vessel since the setpoint would not be based on the coldest fluid temperature.

Address the above concerns by discussing the following:

(1) Show that for all normal events and events in which the RCS fluid temperature is changing, your proposed system suitably protects the reactor vessel at its coldest location.

(2) Show data to justify the RCS temperature transients assumed in (1) above.

(3) Include in your analyses the most limiting single failure, and justify the choice.

(4) Include in your analyses the effects of system and component response times, including;

a. temperature detectors

b. pressure detectors

c. logic circuitry

Show the response times that were assumed and the techniques, including surveillance requirements for ensuring their conservatism.

23. (5.4.7.2.4) Section 5.4.7.2.4 of FSAR states that "Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve designed to relieve the maximum possible back leakage through the valves." What is the basis for determining the maximum possible back leakage? Is this back leakage consistent with a relief flow capacity of 20 gpm at a set pressure of 600 psig? Show that there are design provisions to permit periodic testing for leak tightness of the check valves that isolate the discharge side of the RHRs from the RCS.

24. (5.4.7.2.1) Is there direct position indication for the isolation valves on the suction side of the RHR system?

25. (5.4.7.2.1) The RHR miniflow bypass lines allow bypass flow when RHR pump discharge flow is insufficient. At what frequency is the operability of these miniflow lines verified? What assurances are available to the operating staff that the miniflow isolation valves are not misaligned? Discuss what testing will be performed to validate that the miniflow lines provide an adequate pump flow path such that damage to these pumps will be precluded during this mode of operation.

26. (5.4.7) Branch Technical Position RSB 5-1 specifies in Table 1, Item 1-C that the steam generator atmospheric dump valves (ADVs), their operators, and their power supplies shall be safety grade. FSAR

Section 10.3.1 states that the power operated atmospheric relief valves are part of the safety design basis. Are the ADVs, their operators, and their power supplies considered safety related and therefore are designed to safety grade standards? If not, why not?

27. (5.4.7.2.2.1) Does the RHR pump performance curve take into account instrumentation uncertainties used in deriving the curve?

28. (5.4.7.2.3.4) What precautions and procedures are there that will preclude the cooldown rate from exceeding the Technical Specification limit upon loss of instrument air to both the RHR heat exchanger outlet and bypass flow control valves?

What indications and alarms are available in the control room to inform the operator of a potential excessive cooldown rate and how excessive can this rate become before the situation is turned around?

29. (5.4.7.2.3.5) Section 5.4.7.2.3.5 states that the steam generator power-operated relief valves can be used to attain a primary side cooling rate of 35°F/h. Section 10.3.2.2.3 states that these valves can be used to attain a rate of 50°F/h. Explain this discrepancy. Is the control system used to maintain this rate considered a safety related system and designed to safety grade standards?

30. (5.4.7.2.2.3) The FSAR states that "valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the drain header." Please identify these valves. Do both trains of the RHR system share the same header?

31. (5.4.7.2.3) With the RHR flow high enough such that the miniflow bypass flow valve is closed, is it possible, considering a single failure, that both the residual heat exchanger outlet and bypass flow control valves will be closed? How much time does it take for the miniflow bypass flow valve to open and what is the possible damage to the RHR pump in the interim?

32. (5.4.7.2.3) Discuss the possibilities for air getting trapped in any part of the RHRS during startup and for the air causing water hammer and damage to the RHRS.

33. (5.4.7.1) Provide the calculations of the cooldown times given in Section 5.4.7.1. What values were assumed for the component cooling water temperature, heat transfer surface area and heat transfer coefficient? Show in the calculations that fouling of the heat exchanger was taken into account.

34. (5.4.7.2.3) In Section 5.4.7.2.3.5 it is stated that local manual actions could be performed if permitted by the prevailing environmental

conditions in order to achieve cold shutdown. Would any of these actions become necessary considering a single failure coupled with a loss of either onsite or offsite power and if so what are the actions and where may these actions occur?

35. (5.4.7) Provide detailed information on the sizing criteria used to determine the relief capacity of the RHR system suction line pressure relief valves.

Did the version of the ASME code to which these relief valves were sized require establishing liquid or two-phase relief capacity with testing? If so, describe in detail the test program and results. If the liquid or two-phase relief capacity was not established by test, show that the difference between the rated and maximum required capacity is more than sufficient to bound liquid and two-phase relief rate uncertainties. In the absence of liquid relief valve testing, describe why you believe these valves can reliably pass water without damage to the valve.

36. (5.4.7) Provide additional information regarding the power sources supplied to the RHR isolation valves. The staff's position is that a single failure of a power supply or interlock will not prevent isolation of the RHR when RCS pressure exceeds its design pressure. Additionally, loss of a single power supply cannot result in the inability to initiate at least one 100 percent RHR train.

37. (5.4.7.2.4) What is the design pressure of the RHRS? Section 5.4.7.2.4 states that each RHR relief valve has the capability to maintain the RHRS to within maximum code limits. Identify those design basis events that were excluded from the analysis that determined the relief valve capacity. Provide the bases for the exclusions. Specifically address the capability of the valves to provide relief for the discharge of the charging pumps as well as thermal expansion. Describe the postulated accident events and their sequences, including the discharge of the accumulators, and the combined flow of the safety injection pumps which exceeds the charging pump flow at lower pressures.

38. (5.4.7.2.2.3) Describe the design for the RHRS isolation valves and the tests performed to demonstrate that they will operate properly for the postulated pressure transients and environments.

39. (5.4.7.2.1) What indicates loss of component cooling water to RHR pumps? Do all of these instruments meet IEEE 279 requirements? How long could the pumps continue to run following a loss of component cooling water without damage? Provide date to support your response

40. (5.4.7) Provide or reference a discussion of your compliance with each item of RSB BTP 5-1 in NUREG-0800. Justify any deviations from this Branch Technical Position.

It is the staff's position that all operator actions necessary to take the plant from normal operation to cold shutdown should be performed from the control room. Indicate whether there are any systems or components needed for shutdown cooling which are de-energized or have power locked out during plant operation. If so, indicate what actions have to be taken to restore operability to the components or systems. In particular, address the accumulation system.

41. (5.4.7.2.2) In the event the RHS relief valves open, describe the means available to alert the operator of the situation. Verify that procedures will be available to the operator for responding to this event.

42. (5.4.7) Provide the following information related to pipe break or leaks in high or moderate energy lines outside containment associated with the RHR system when the plant is in a shutdown cooling mode:

1. Determine the time available for operator action based on the maximum discharge rate from a pipe break in the systems outside containment used to maintain core cooling.

2. Describe the alarms available to alert the operator to the event, the recovery procedures to be utilized by the operator.

43. (5.4.7) Recent plant experience has identified a potential problem regarding the loss of shutdown cooling during certain reactor coolant system maintenance evaluations. On a number of occasions when the reactor coolant system has been partially drained, improper reactor coolant system level control, a partial loss of reactor coolant inventory, or operating the RHR system at an inadequate NPSH has resulted in air binding of the RHR pumps with a subsequent loss of shutdown cooling. Regarding this potential problem, provide the following additional information.

1. Discuss the design or procedural provisions incorporated to maintain adequate reactor coolant system inventory, level control, and NPSH during all operations in which RHR cooling is required.

2. Discuss the provisions incorporated to ensure the rapid restoration of the RHR system to service in the event that the RHR pumps become air bound.

3. Discuss the provisions incorporated to provide alternate methods of shutdown cooling in the event of loss of RHR cooling during shutdown maintenance. These provisions should consider maintenance periods during which more than one cooling system may be unavailable, such as loss of steam generators when the reactor coolant system has been partially drained for steam generator inspection or maintenance.

44. (5.4.7.2) Describe the consequences of a failure associated with the isolation valves in the suction line from the hot leg to the RHR pumps during normal shutdown cooling. Evaluate this event assuming that only one RHR train is operating at the time of the failure. Describe the consequences of this event assuming (a) the reactor vessel is closed, and (b) the reactor vessel head has been unbolted. The failure could be caused by operator error or a passive failure such as the gate separating from the stem. These failures could cause pump damage due to cavitation and loss of core cooling. Discuss the operator actions required to mitigate the consequences, describe the alarms available to alert him to the situation and the time frame available to perform the required action.

45. (5.4.7.2.3) Describe your proposed program for verification of adequate mixing of borated water added to the RCS under natural circulation conditions and confirmation of natural circulation cooldown ability, in accordance with the criteria of BTP RSB 5-1.

46. (5.4.13) The FSAR states that the PORVs provide a safety related means for RCS depressurization to achieve cold shutdown. Does this mean that the entire PORV system is designed to safety related criteria?

47. (5.4.15) The FSAR description of the Reactor Vessel Head Vent system (RVHVS) is incomplete. Please provide the following information to show compliance with the requirements of Action Item II.B.1.

(1) Please amend the RCS P&ID (Figure 5.1.2-1) to show the piping, valves, and instrumentation, including control and indication, for the RVHVS. Include in this drawing the identification associated with these items.

(2) Describe the means for venting the pressurizer. Describe the procedures that would assure sufficient coolant can enter the U-tube region so that sufficient decay heat can be removed from the RCS.

(3) Describe RVHVS testability.

(4) Describe the position indication for the RVHVS valves that is available in the control room.

(5) Describe the control system that operates the RVHVS valves. Describe its compliance with safety related requirements.

(6) Identify the power sources for the modulating valves.

(7) Describe the capability of the system to vent the RCS hot and cold legs.

(8) Provide a reliability analysis consisting of a failure mode and effects analysis (FMEA) or equivalent qualitative analysis that shows that no single active component failure, human error, or test and maintenance action could result in inadvertent opening or failure to close after intentional opening of an RCS vent path. Include in the analysis components in the associated power, instrumentation, and control systems as well as the electrical and mechanical components of the RCS vent system (reference NUREG-0737 Item II.B.1 Clarification A.(7) and (8)).

48. (5.2.2) Will the PORV setpoints be adjusted over time for low temperature overpressure protection in order to account for vessel embrittlement? Justify your response.

(8) Provide a reliability analysis consisting of a failure mode and effects analysis (FMEA) or equivalent qualitative analysis that shows that no single active component failure, human error, or test and maintenance action could result in inadvertent opening or failure to close after intentional opening of an RCS vent path. Include in the analysis components in the associated power, instrumentation, and control systems as well as the electrical and mechanical components of the RCS vent system (reference NUREG-0737 Item II.B.1 Clarification A.(7) and (8)).

48. (5.2.2) Will the PORV setpoints be adjusted over time for low temperature overpressure protection in order to account for vessel embrittlement? Justify your response.