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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 the final set of questions concerning the Vogtle plant. These questions are primarily a result of a review of those sections of Chapter 15 of the FSAR for which Reactor Systems Branch has primary review responsibility.

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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REQUEST FOR ADDITION INFORMATION

GEORGIA POWER CORPORATION

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 424/425

440.76 (15.0) Section 15.0.8 states "The pressurizer heaters are not assumed to be energized during any of the chapter 15 events". For each of these events show that this is a conservative assumption or quantify the effects of the heaters being energized.

440.77 (15.0) Section 15.0.8 states that "A control system setpoint study will be performed prior to operation to simulate performance of the reactor control and protection systems. In this study, emphasis is placed on the development of a control system which will automatically maintain prescribed conditions in the plant even under the most adverse set of anticipated plant operating transients with respect to both system stability and equipment performance". Show that the results of this study and the system setpoints are consistent with the accident analysis assumptions and that these assumptions are conservative taking into consideration instrumentation errors.

440.78 (15.0) A change in the Westinghouse fuel rod internal pressure design criteria will permit the internal fuel rod pressure to exceed system pressure. For some events, this will result in an increase in the number of rods normally predicted to fail. If the fuel design is based on this higher fuel rod internal pressure design criteria, show that the effects of the higher fuel rod internal pressure have been properly factored into predictions of the effects of fuel rod ballooning and number of rod failures.

440.79 (15.0) Discuss the loss of instrument air showing that it meets the appropriate acceptance criteria for a moderate frequency event. Causes of a loss of instrument air and consequences should be addressed. Include in the discussion any instructions given to the operator to place the plant in a safe condition and any alarms and indications that the operator would have to rely upon. The loss of instrument air should be considered during all phases of reactor operation. Also, present your plans and capability for preoperational or startup tests to substantiate the analyses.

440.80 (15.0) How were the operator action times assumed in the Chapter 15 analyses established? Do these times agree with those stated in ANSI N660? If not, please justify the times assumed.

Describe the operator actions that are required to mitigate the consequences of a boron dilution event during the various modes of operation. Include a discussion of what instrumentation and alarms will alert the operator to the event. Will the operator still be alerted in the event of a single failure?

For the boron dilution event and for those accidents noted in 15.0.13 for which operator action is required, what would be the impact of no operator action or a closely related but erroneous action?

440.81 (15.0) Table 15.0.8-1 specifies plant systems and equipment available for transient and accident conditions. The list appears to be incomplete for some conditions. For example, Table 15.1.5-1 lists the required equipment following a rupture of a main steam line. This list includes the RHRS and containment sprays which are omitted from Table 15.0.8-1. Please amend Table 15.0.8-1 to reflect a complete tabulation of required systems and equipment for those transients and accidents described in Chapter 15.

440.82 (15.0) Show that incidents of moderate frequency that are analyzed in Chapter 15.0, including the complete loss of forced reactor coolant flow accident, would not generate a more serious plant condition without other faults occurring independently. Section 15.0.1.2, in discussing Category II events, states that "By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV events". Loss of nonemergency AC power to the station auxiliaries (Section 15.2.6) is defined as a Condition II event. In Section 15.3.2 loss of power to the RCS pumps is the initiator of the complete loss of forced reactor coolant flow, which is classified as a Condition III event. This should be classified as a Condition II event. Show that this transient meets the Condition II criteria.

440.83 (15.0) What are the initiation and completion of action times of the ECCS components that were used in the Chapter 15 analysis with and without offsite power? What are the bases for these times? Provide verification that the valve discharge rates and response times (such as opening and closing times for main feedwater, auxiliary feedwater, turbine and main steam isolation valves and steam generator and pressurizer relief and safety valves) have been conservatively modeled in the Chapter 15.0 analyses.

440.84 Provide as part of Table 15.0.3-2, or where appropriate, the initial pressurizer water volume assumed in applicable Chapter 15 accident analyses. Include a discussion to indicate the degree of conservatism provided by the pressurizer volume assumed. Will the assumed initial pressurizer level be a technical specification limit? If not, why not?

440.85 (15.0) Summary block diagrams similar to those provided for other events, should be provided for the following events:

Turbine trip (subsection 15.2.3)

Inadvertent closure of main steam isolation valves (subsection 15.2.4)

Loss of condenser vacuum and other events resulting in turbine trip (subsection 15.2.5)

Steam system piping failure (subsection 15.1.5)

Inadvertent loading and operation of a fuel assembly in an improper position (subsection 15.4.7)

Radioactive liquid waste system leak or failure (subsection 15.7.2)

440.86 (15.0) The following pertain to Chapter 15 Event Block Diagram Sequences found in Section 15.0.1 of the FSAR.

1. The block diagram sequence for the Dropped Rod Cluster Control Assembly (Figure 15.0.1-10) includes a reactor trip from full power. Normally, the turbine is tripped automatically on reactor trip, so either the turbine bypass system, or power operated relief valves, or safety relief valves must be actuated to handle steam from the steam generators. Since only safety grade systems are assumed to operate during the transient, the safety relief valves would be assumed to operate. They should thus be shown in the diagram.

2. Figure 15.0.1-8 "Loss of Forced Reactor Coolant Flow."

a. For the partial loss of flow and single pump locked rotor, Item 1 applies.

b. For total loss of flow, since offsite power is assumed to be lost, the main feedwater pumps would be lost and the auxiliary feedwater would be required. A sequence for auxiliary feedwater should therefore be shown on the sequence diagram.

3. Figure 15.0.1-12, the analysis of this event has assumed maximum permissible power with one loop out of service. This leads to the potential requirement for the secondary safety relief valves. Refer to Item 1 for discussion.

4. Figure 15.0.1-3, "Depressurization of Main Steam System."

Since the main steam lines will be isolated during this transient, the secondary safety relief valves will probably be required for heat removal from the secondary system. If this could occur, a sequence for secondary relief valve actuation should be added to the diagram.

5. Figure 15.0.1-11, "Single Rod Cluster Control Assembly Withdrawal at Full Power."

See Item 1 for discussion of the possible need for secondary safety valve actuation on reactor trip from full power.

6. Figure 15.0.1-7, "Major Rupture of a Main Feedwater Line." Same comment as Item 5.

7. Figure 15.0.1-14, "Rupture of a Control Rod Drive Mechanism Housing," Same comment as Item 5.

8. The Chapter 15 event diagrams that have assumed turbine trip or reactor trip in the analyses should include a sequence for turbine trip, with appropriate single failure designations.

440.87 (15.1.5) Provide justification for the assumed core flow during a major steam line break in accordance with the Standard Review Plan (SRP 15.1.5) Acceptance Criteria.

440.88 (15.15) The steam system piping failure with loss of offsite power (LOOP) assumes a main steam line break coincident with the LOOP. On recent applications, we have been allowing the LOOP to be initiated by the turbine trip. Show that assuming the LOOP at event initiation is more limiting than assuming LOOP as a result of turbine trip for piping failures of varying sizes

440.89 (15.1.5) Provide more detailed information concerning the auxiliary feed system and operator action assumed for the main steam line rupture analysis. Specifically address:

1. Assumed auxiliary feed flow

2. Time to deliver auxiliary feed

3. Auxiliary feed temperature

4. Operator actions assumed

5. Time frame for operator action

6. Alarms and indications provided to assist the operator in determining the correct course of action.

440.90 (15.1.4, 15.1.5) Figures 15.1.4-3 and 15.1.5-2 of FSAR indicate that the pressurizer is emptied during the inadvertent opening of steam relief or safety valves and steam line break transients. Discuss the potential effects of this condition, including the potential for and recovery from void formation in the RCS.

For many events analyzed, voiding in the primary system is expected to occur. Confirm that the plant operators have been instructed in:

a. understanding that voiding can and may occur

b. recognizing the symptoms of voiding

c. the significance of voiding on plant performance

d. steps to avoid voiding and methods to control and eliminate voiding should it occur.

What simulator training will the operator receive that adequately simulates the voiding process? If you do not intend to use a simulator that can adequately predict voiding, justify why this is acceptable in light of the extensive operating experience which indicates operators still do not know how to handle voiding.

440.91 (15.1.4) Provide the minimum DNBR vs. time curve for the inadvertent opening of a steam generator relief or safety valve event.

440.92 (15.1) General Design Criterion 17 states that specified acceptable fuel design limits (SAFDLs) must be met for anticipated operational occurrences and that unacceptable fuel failures (e.g., doses exceed 10 CFR 100 values) should not occur for postulated accidents, assuming offsite power is not available. Please demonstrate that for all of the anticipated operational occurrences (AOOs) and postulated accidents (PAs) analyzed in Chapter 15, these limits are still met assuming loss of offsite power. Note that assuming loss of offsite power per GDC-17 is not considered to satisfy the single failure criterion.

440.93 (15.1.5) Confirm that the time of core life was chosen to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile and radial power distribution for the steam line break event.

440.94 (15.1.5) In section 15.0.13 of the FSAR, it is stated that for a steam line break upstream of the MSIVs, that the operator can evaluate which is the affected steam generator within one minute and isolate it from auxiliary feedwater. Justify the acceptability of the one minute action time assumed. Provide all operational or simulator data which supports your assumption. Explain how operator error or delay is accounted for. In the absence of supporting experimental data, assumed operator action times should be consistent with ANSI N660 and no less than 10 minutes.

440.95 (15.1.5) The main steam line rupture analysis assumes zero power in order to arrive at the most limiting cooldown transient. This assumption however may not be conservative when analyzing the event from a DNBR point of new. Please analyze this event at full power or show that the DNBR is bounded by the zero power cases.

440.96 (15.2.6) Table 15.2.3-1 indicates that the reactor coolant pumps begin coastdown at 61 seconds following the loss of non-emergency AC power. Provide justification as to why the value was selected and not any other time, including t=0. Describe any reliance upon non-safety related equipment and/or operator action to achieve this time. What would be the effect on the time variations of the minimum DNBR, heat flux (average and maximum), primary system pressure, core average temperature and pressurizer volume as a function of time for the case where coastdown begins at t=0?

440.97 (15.2.6, 15.2.7) Provide the variations over time of the minimum DNBR, neutron power, heat flux (average and maximum) and coolant exit temperatures (average and hot channel) for the loss of AC power and normal feedwater events. Specify the number of fuel rods, if any, expected to be in DNB.

440.98 (15.2.6) Provide the basis for the steam generator heat transfer coefficient and flow during natural circulation flow in the RCS. Describe the available data, or data that you will obtain, which will verify the acceptability of the analysis of the loss of nonemergency AC power accident.

440.99 (15.2.6, 15.2.7) For the loss of AC power and normal feedwater transients, discuss the reactivity coefficient assumptions, and show that the power response and reactivity coefficients used in the analysis are conservative.

440.100 (15.2.6, 15.2.7) Table 15.2.3-1 indicates that the main feedwater flow stops at 10 seconds for the loss of AC and normal feedwater transients. Provide justification for the selection of this value. Describe any reliance on non-safety related equipment or manual actions to achieve this value. Describe the effect of losing the feedwater at t=o. Include in the discussion the variations over time of the minimum DNBR, heat flux (average and maximum), primary system pressure, pressurizer volume, core average temperature and exit temperatures.

440.101 (15.2.6, 15.2.8) Describe the assumptions made for the loss of AC power and normal feedwater flow transients with respect to the scram characteristics, i.e. time delay for rods to drop and those rods not dropped into the core. Describe any credit taken for the functioning of normally operating plant systems.

440.102 (15.2.7) Section 15.2.7 notes that a reanalysis of the loss of normal feedwater event will be provided to address the interaction between control and protection systems. Provide this analysis.

440.103 (15.2.7) In the loss of normal feedwater event, was credit taken for manually tripping the reactor coolant pumps? If so, describe the procedures, alarms and indications that aid the operator to take action. At what point into the transient would this action be taken?

440.104 (15.2.8) Section 15.2.8.2.1 states that "the auxiliary feedwater motor-driven pump delivers 510 gal/min to the three intact steam generators". Shouldn't the flow come from both motor-driven pumps? If the flow is from only one pump, as stated, the only way to direct flow to three steam generators is to open two locked closed valves in the interconnect line. If this is the case, please describe the amount of time available to the operator to perform this function and the procedures and control room indications available to aid the operator. How much time is assumed in the analyses for the actions to be taken?

440.105 (15.2.8) Give a qualitative description of the trends shown by the curves provided in 15.2.8. Include in this discussion the following points:

1. Figure 15.2.8-2 shows pressurizer water volume holding steady at 1900 ft^3, Table 5.4.10-1 denotes pressurizer volume to be 1800 ft^3 and section 15.2.8.2.2 states that water is not relieved from the pressurizer for the main feedwater line break event with offsite power. Please explain this discrepancy. If water is in fact relieved through the pressurizer safety valves, provide justification for the water relief rate assumed in the analysis, and confirm that the safety valves are designed for liquid relief. If not, justify why they should not be and explain why you did not assume them to remain stuck open.

2. Figure 15.2.8-3 shows virtually identical hot and cold leg temperatures - thus it would seem that the intact steam generators are not removing heat from the primary (even when the hot leg tempera-

tures are above the saturation temperature for 1250 psi, the design pressure for the steam generators). Please explain.

3. Figure 15.2.8-2 shows pressurizer pressure decreasing at the same time the pressurizer appears to be relieving water. Please explain.

440.106 (15.2.8) For the feedwater line break, provide the variations over time for the minimum DNBR, discharge rate through the break, steamline and feedwater flow rates and safety and relief valve flow rates. Discuss the extent of fuel damage.

440.107 (15.2.8) Show that the initial core flow assumed for the analysis of the feedwater line rupture event was chosen conservatively.

440.108 (15.2.8) Provide a detailed discussion, supported by sensitivities studies, that shows the most limiting combinations of reactivity coefficients, power profiles, core flow, rod worths (including the maximum worth rod is in the fully withdrawn condition), safety injection flow, etc., have been evaluated to identify the worst case responses such as the most limiting combination for minimizing DNBR or the most limiting combination for maximizing the effects of a return to power following reactor trip.

440.109 (15.3.3) Figure 15.3.3-1 (locked rotor event) indicates that the faulted loop flow becomes negative in under one second and reaches about-35% of nominal flow in about two seconds. Please explain these phenomena.

440.110 (15.3.3) Provide a figure showing DNBR vs time for the locked rotor event. What fraction of the fuel rods were assumed to fail for this event?

440.111 (15.3.3) For the locked rotor event, what assumptions were used in the analysis for the reactivity coefficients and the axial and radial power distributions. Demonstrate these values form the most limiting combination.

Verify that conservative scram characteristics were assumed in the analysis, i.e., maximum time delay with the most reactive rod held out of the core.

440.112 (15.4.4) For the startup of an inactive reactor coolant pump at an incorrect temperature event, describe the analysis assumptions regarding the allowance used to account for power measurement uncertainty, the axial and radial power distributions and the scram characteristics.

The analysis also assumes that the idle pump will reverse the flow in the loop and achieve a nominal full-flow condition in approximately 20 seconds. Please describe how and when this will be verified. (15.4.4.2.1)

440.113 (15.4.6) The only results in the FSAR for the boron dilution event during the various modes of operation are the times available to the operator to manually terminate the source of dilution flow. Please provide the temporal variations of the core reactivity, DNBR and power level.

440.114 (15.4.4) The analysis for startup of an inactive reactor coolant pump at an incorrect temperature assumes an initial condition of steady state power of about 70% (Figure 15.4.4-1). According to the Technical Specifications for previous Westinghouse plants (3.4.1.1), plant operation at 70% power with only 3 pumps running is permitted as long as the hot standby condition is attained within one hour. In order to be in such a situation, either the plant is coming down in power, to achieve hot standby, and is therefore not in a steady state condition or the plant is operating in the n-1 loop configuration.

If the former is the case, show that the analysis assumption of steady state operation is conservative. If the latter is the case, then additional information will have to be provided concerning plant operation and analyses including, but not limited:

1. Meeting Chapter 15 acceptance criteria with N-1 loop operation

2. P&IDs showing primary side and secondary side valve alignments including the main and auxiliary feedwater and main steam systems

3. The effects on core thermal hydraulics due to asymmetric flow

4. Loop seal injection

5. The consequences of N-1 loop operation on generic issues such as water hammer and pressurized thermal shock

6. The effects of N-1 loop operation on auxiliary systems such as pressurizer spray

7. A description of the fluid (temperature, pressure and flow) in the inactive loop and its associated steam generator under all conditions

8. The effects of N-1 loop operation on the capability to provide adequate safety-grade decay heat removal capability

9. The effects of reverse flow in the inactive loop since Vogtle does not have loop isolation valves.

440.115 (15.4.6) Sections 15.4.6.2.1.2, 15.4.6.2.1.3 and 15.4.6.2.1.4 state that "dilution flow is assumed to be the combined capacity of the two primary water makeup pumps (approximately 242 gal/min)." However, section 9.2.7.2.2 states that each makeup pump is rated at 200 gpm and a head of 285 ft. Please resolve this discrepancy.

Furthermore, at nominal operating pressures, the charging pumps are each capable of flowrates well in excess of 200 gpm. As a result, the suction side of the charging pumps and therefore the head of the primary makeup pumps may drop below 285 feet. Please determine what the head and discharge rate of the makeup pumps will be. (15.4.5.2.1)

440.116 (15.4.6) The FSAR states that the boron dilution event will be precluded from occurring during refueling because certain valves will be locked closed. Will power also be removed from these valves and will these conditions be placed in the plant Technical Specifications? Justify why administrative controls are sufficient and why operator error won't occur. Describe the effects and consequences of single failures and operator errors. Justify the assumption that the only source of unborated water is isolated by closure of these valves. Boron dilution events have occurred by backflow from leaking steam generators during conditions when the secondary pressure was above the primary pressure. Evaluate this condition or show how you will preclude it from occurring (i.e., tech spec limit on secondary to primary pressure difference).

440.117 (15.4.6) Are there two independent boron dilution alarms in order to demonstrate that the single failure criterion is met?

440.118 (15.4.6) As reactor conditions change (i.e., neutron source decay, moderator temperature change, or control rod position changes), the boron dilution alarm setpoint will need to be adjusted. Will this be an automatic or manual adjustment at Vogtle? If manual, what is the frequency of adjustment and does the setpoint methodology take into account the uncertainties provided by the changing reactor conditions? What provisions have you made to assure that boron dilution alarms cannot be taken out of service?

440.119 (15.4.6) As noted in FSAR section 15.4.6.4, the VEGP is not in compliance with the SRP 15 minute margin to terminate the boron dilution event for hot standby and cold shutdown conditions. Please describe how you will comply with this criterion.

440.120 (15.4.6) Reference or describe the analytical model used for obtaining the results in Section 15.4.6.2. Discuss the degree of conservatism incorporated in this model.

440.121 (15.4.6) A PWR recently experienced a boron dilution incident due to inadvertent injection of NaOH into the reactor coolant system while the reactor was in a cold shutdown condition. Discuss the potential for a boron dilution event caused by the chemical addition portion of the CVCS and by dilution sources other than the CVCS (for example, via the engineered safety systems).

440.122 (15.4.6) For the boron dilution event, please discuss the VEGP analysis and demonstrate how the following criteria were met:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values. In particular, consider the case of a dilution event occurring while the reactor vessel head is on and the system is in a water solid condition.

2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs.

3. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, the number of fuel failures must be assumed for all rods for which the DNBR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

4. For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2% to account for power measurement uncertainty.

5. The core burnup and corresponding boron concentration are selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coeffi-

cient, axial power profile, and radial power distribution. This will usually be the beginning-of-life (BOL) condition.

6. All fuel assemblies are installed in the core.

7. For each event analyzed, a conservative high reactivity addition rate is assumed taking into account the effect of increasing boron worth with dilution.

8. Conservative scram characteristics are assumed, i.e., maximum time delay with the most reactive rod held out of the core.

440.123 (15.5.1) It is stated in the FSAR for the inadvertent operation of the ECCS event that the operator is to determine whether the SI signal is spurious or steady state and to decide whether to block the signal. Provide the criteria that the operator will use in making the decisions. (15.5.1.1)

440.124 (15.5.1) In Section 15.5.1.2 of the FSAR the assumption of zero injection line purge volume (initial injection is borated to 2000 ppm) is used. For this assumption to be valid, you must either commit to having the injection line always filled to 2000 ppm borated water or provide a discussion as to why this is more limiting than the possible transient that could be caused by the injection of unborated cold water into the cold legs.

440.125 (15.6.1) Describe Vogtle compliance to the requirements of generic letter 83-10c. Show that the assumptions made regarding reactor coolant pump operation for your transient and accident analyses are conservative with respect to the expected RCP operation which will result from resolution this generic letter. This generic letter presents the staff resolution of TMI Action Plan Item II.K.3.5, therefore section 5.4.1.1 of the FSAR should be modified accordingly.

440.126 (15.6.3) Clarify whether you analyzed a case which considers the radiological effects of a steam generator tube rupture with the highest worth control rod stuck out of the core.

440.127 (15.6.1) Describe the recovery from an inadvertent opening of a pressurizer safety valve accident. Include information on operator action, pressurizer water level, potential for void formation in the RCS, hot and cold leg temperatures and core flow. The FSAR states that there is an initial rapid decrease in the RCS pressure until this pressure reaches the hot leg saturation pressure, at which point the decrease is slowed considerably. At what point in time does this occur since it is not evident in the curve of pressurizer pressure vs time. Also describe the possibility of void formation in the hot leg and any resultant decrease in heat transfer to the secondary. (15.6.1.2)

440.128 (15.6.1) Does the analysis for the inadvertent opening of a pressurizer safety valve take into account tripping of the reactor coolant pumps? If so, at what time? (15.6.1.2)

440.129 (15.6.3) Section 15.6.3.1 of the FSAR states "charging pump flow increases in an attempt to maintain pressurizer level" and "feedwater flow to the affected steam generator is reduced as a result of primary coolant break flow to that unit". Are any control systems used to maintain these levels in the analysis? If so, justify that their operation which you have assumed is conservative and modify Table 15.0.8-1 to include them.

Has credit been taken for the steam generator blowdown liquid monitor or the condenser air ejector radiation monitor? If so, modify Table 15.0.8-1. (15.6.3.1)

440.130 (15.6.3) In Section 15.0.1 and 15.6.3.1 the steam generator tube rupture event is stated to be an ANS Condition IV event. This is an event not expected to take place but is postulated because of its potential to have significant amounts of radioactive material released. In view of the occurrence of the SGTR event at Ginna, among others, how can this be classified as an event that will not occur over the life of the Vogtle plant? Either justify the event as a Condition IV event or categorize it to a condition commensurate with operating experience.

440.131 (15.6.3) Figures 15.6.3-1 and 15.6.3-4 show a differential pressure of about 1000psi between the primary and faulted steam generator at 30 minutes. Figure 15.6.3-11 shows an increasing water volume due to the break flow rate as shown in Figure 15.6.3-9 at 30 minutes. Section 15.6.3.2.1 states, however, that leakage flow through the ruptured tube is assumed to be terminated with 30 minutes of the initiation of the event. Unless these parameters show discontinuous behavior at 30 minutes, it would appear that the assumption and the figures are in conflict. Please resolve this.

If leakage flow is terminated at 30 minutes, how is it accomplished? Any equipment used should be listed in Table 15.0.9-1 and qualified.

If the leakage flow is not terminated at 30 minutes and since the flow through the steam generator safety valve has approached a non-zero asymptote at this time, it would appear that additional radioactive material will be released to the atmosphere. In this event, you will need to reanalyze the radiological consequences.

440.132 (15.6.3) It is stated in Section 15.6.3.1 of the FSAR that given the control room indications and the magnitude of the break flow, that the accident diagnostics and isolation procedure can be completed within 30 minutes of mitigation of the event. Recent SGTR events of Ginna, Point Beach and Praire Island indicate that the release from the effected steam generator takes place for over 30 minutes. Therefore, to fully evaluate this event analysis:

(1) Submit an evaluation of operator actions necessary to effect pressure equalization, and a conservative time estimate for each action, as well as initial delay time. Consider that these actions may have to be achieved under loss-of-offsite power/natural circulation conditions under which a steam bubble might form in the reactor vessel head.

(2) Discuss: (a) whether, as a result of possible modification of its analysis, including consideration of longer leak times, liquid can enter the main steamlines, and (b) what would the effects be on the integrity of the steam piping and supports, considering both the liquid dead weight and the possibility of water hammer. Unless the applicant can demonstrate that the incident will be terminated within a time period sufficiently short to avoid steam generator overfill, the applicant should submit the results of an analysis that demonstrates that the integrity of the steamlines and supports will be maintained.

(3) Verify that any components that are credited in the analysis to mitigate the consequences of the SGTR, including the motive power sources,

are classified as safety related; meet applicable GDCs, including GDCs 1, 2 and 4; are seismically and environmentally qualified; and have sufficient capability to equalize primary and secondary pressure within the time period postulated in the response to items (1) and (2) above. If any components which do not meet the above requirements are relied upon to mitigate the SGTR accident, then a justification should be provided for taking credit for the proper operability of such components.

(4) Provide the noding diagram used in the analysis. Justify that sufficient noding is provided to predict head bubble formation or loss of natural circulation in loops for which the steam and feedwater flow has been isolated.

(5) Provide the most limiting single active failure. If the most limiting single active failure is failure of an atmospheric relief valve to close, operator action to close the block valve may be assumed if justified.

440.133 (15.6.5) Confirm, that during the reflood stage, that the Vogtle analysis for the LOCA event conforms to 10 CFR Part 50 Appendix K whereas the reactor coolant pumps should be assumed to have locked impellers if this assumption leads to maximum cladding temperature, otherwise the rotor is assumed to be running free.

440.134 (15.6.5) The LOCA analysis presented in 15.6.5 shows the results for discharge coefficients 0.4, 0.6 and 0.8. Appendix K requires calculations for discharge coefficients up to 1.0. Either provide new analyses for CD=1.0 or confirm that it is not the limiting case for the spectrum of break sizes.

440.135 (15.6.5) Does the LOCA model include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and outside of the cladding? It so, identify justifying documentation. If not, correct the LOCA analyses to include adequate treatment of fuel cladding and rupture.

440.136 (15.6.5) Identify single failures and operator errors that would divert ECCS flow. For both large and small breaks discuss the effect of these failures on flow to the core, the containment water level and conformance with the 10 CFR 50.46 acceptance criteria.

440.137 (15.6.5) In the LOCA analysis, an upper head temperature equal to the cold leg temperature is assumed. Justify this assumption.

440.138 (15.6.5) Provide an analysis of the transient resulting from a break in the ECCS injection line. Describe the flow splitting which will occur in the event of the most limiting single failure and verify that the amount of flow actually reaching the core is consistent with the assumptions used in the analysis. Show that 10 CFR 50.46 acceptance criteria are satisfied.

440.139 (15.6.5) Figure 15.6.5-36 shows safety injection flow increasing to a constant value of about 20 lbs/sec (150 gpm) at minimal core pressures (Figure 15.6.5-11, for times greater than 30 seconds). However, the SI pump performance curve (Figure 6.3.2-5) shows flow from each pump to be in excess of 700 gpm for these pressures. Please explain the difference. If it is assumed that some of the SI flow is lost through the break, what is the justification for determining the amount lost? Does Figure 15.6.5-36 include flow from one or two pumps?

440.140 (6.3) The response to satisfying the requirements of TMI Action Item II.K.3.10 is inadequate. Please demonstrate that for reactor trip on turbine trip, for power levels above the P-9 setpoint, the probability of a small break LOCA resulting from a stuck open PORV is substantially unaffected. (7.2.1.1.2)

440.141 (6.3) TMI Action Item II.K.1.10 requires that you are to have "procedures for removing safety related systems from service (and restoring to service) to assure operability status is known". Section 13.5 of the FSAR states that these procedures will be in place. Are these procedures now written? If not, commit to having these procedures in place prior to initial fuel load. (13.5.1.2)

440.142 (5.2.2) The response to TMI Action Item II.D.3 is incomplete. Please describe the kind of instrumentation that is provided, whether or not the indication is in the control room, whether the indicated information has been integrated into procedures and training, and what alarms are associated with the indication. Will the valve position indication be seismically qualified and safety related? If not, why not? (5.4.13.2)