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March 27, 1991

ELV-02593
0848

Docket Nos. 50-424
50-425

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
RESPONSE TO NRC QUESTIONS ON
VANTAGE 5 FUEL SUBMITTAL

On November 29, 1990, Georgia Power Company (GPC) submitted letter ELV-02166 which requested changes to the Technical Specifications associated with the proposed use of VANTAGE 5 fuel. Additional information related to that request was submitted to the NRC via letter ELV-02363 dated January 29, 1991. On February 13, 1991, a meeting was conducted with the NRC to discuss the previous submittals. As a result of that meeting, the NRC identified 16 requests for additional information.

Attached to this letter are responses to the 16 questions that were attached to the NRC record of the February 13 meeting. It should be noted that the responses include references to three other transmittals. The first was provided with letter ELV-02597 dated March 6, 1991, which transmitted the proprietary documents requested by the NRC. The second transmittal will be an additional Technical Specifications change request to eliminate the resistance temperature detector bypass manifolds. The third transmittal will include a revised Technical Specification to account for an increase in reactor coolant system flow uncertainty due to the uncertainty in feedwater flow measurements with venturi fouling.

The second and third transmittals are scheduled to be submitted by the end of March 1991. As stated at the meeting on February 13, GPC expects to begin fuel delivery of the VANTAGE 5 fuel in June 1991. In order to support the scheduled refueling outage, GPC requests that the appropriate Technical Specifications changes be approved by June 1991.

Sincerely,

C.K. McCoy

C. K. McCoy

CKM/HWM/gmb
Attachment

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xc: Georgia Power Company
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U. S. Nuclear Regulatory Commission
Mr. S. D. Ebneter, Regional Administrator
Mr. D. S. Hood, Licensing Project Manager, NRR
Mr. B. R. Bonser, Senior Resident Inspector, Vogtle

NRC VANTAGE 5 SUBMITTAL QUESTIONS

1. A proposed methodology change would allow for control rod insertion following a large break LOCA by taking credit for leak before break. The staff does not support the use of the leak before break methodology with respect to Chapter 15 events. Therefore, RWST boron concentration must be relied upon to maintain subcriticality at cold conditions during post LOCA conditions, rather than control rod insertion.

   Response

   The proposed use of leak before break (LBB) methodology discussed in section 5.2.5 of enclosure 4 to letter ELV-02166 is only applied to the verification that the reactor core will remain subcritical following a large break LOCA. This verification is made prior to each fuel cycle. The method for performing the verification is outlined in Technical Specification Bases section 3/4.5.4. The proposed changes to this Bases section, contained in the VANTAGE-5 submittal, allow credit for the control rods when performing the verification. None of the other Technical Specifications changes or analyses presented with letter ELV-02166 depend on LBB methodology.

   The proposed use of LBB is to support taking credit for the control rods when verifying that the reactor will remain subcritical during the long term cooling phase, following a large break LOCA. The analysis of Chapter 15 events will continue to be performed in the same manner whether or not credit is taken for control rods. Credit for the control rods is allowed by General Design Criterion 27. Credit for the control rods when verifying that the reactor will remain subcritical during the long term cooling phase following a small break LOCA has already been accepted by the NRC. In order to assure that the control rods will be inserted during the long term cooling phase following a large break LOCA it is necessary to confirm that the dynamic effects of the large break LOCA will not prevent subsequent rod insertion. General Design Criterion 4 of 10 CFR 50 Appendix A allows the dynamic effects of certain postulated piping ruptures to be excluded when analyses reviewed and approved by the NRC demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. This demonstration has been previously made using the LBB methodology, and accepted by the NRC for the VEGP reactor coolant loop piping, as documented in FSAR section 3.6. Since the NRC has already accepted the insertion of control rods for small breaks less than 1 $ft^2$ and this application of LBB excludes the dynamic effects of breaks in excess of 1 $ft^2$, GPC has concluded that the dynamic effects of the postulated pipe rupture will not prevent control rods from being in the reactor during the long term cooling phase following any credible reactor coolant system piping rupture.

   Paragraph 3.9.N.2.5.2.2 of the FSAR states that all but four of the control rods would be expected to function following the largest size break. The use of LBB in conjunction with GDC 4 allows the dynamic effects of the largest size breaks to be excluded. The application of LBB has already been approved by the NRC for the exclusion of the dynamic

effects of pipe breaks relative to other components. By applying the LBB logic to exclude the dynamic effects of pipe breaks on the control rods, the verification of subcriticality for long term cooling following a large break LOCA can be done in a similar fashion as for the small break LOCA. Otherwise, the application of paragraph 3.9.N.2.5.2.2 would indicate that the credit for control rods for large break LOCAs should exclude the four control rods that could be affected by a large break.

This application of LBB only affects the bases for the method of verifying that the Refueling Water Storage Tank contains sufficient boron to assure that the reactor will remain subcritical following a large break LOCA. It affects this bases by allowing credit for the control rods. This application of LBB does not have any other effects on the methodology for complying with the requirements of 10 CFR 50.46 and 10 CFR 50 Appendix K.

The RWST boron concentration Technical Specifications limits were previously increased from 2000 - 2200 ppm to 2400 - 2600 ppm in 1988. In order to accommodate the higher capacity and longer fuel cycles, GPC anticipates that future cycles would eventually require a further increase in the Refueling Water Storage Tank (RWST) boron concentration unless credit is taken for the effects of the control rods. Such an increase would involve extensive reanalyses and additional changes associated with increases in calculated doses, equipment environmental qualification, and potential design changes to mitigate the problems associated with higher boron concentrations. These reanalyses would involve reducing the current analysis margins for post accident doses as identified in Table 15.6.5-6 of the FSAR. A further increase in the boron concentration would result in a real reduction in the iodine decontamination factor available from the containment spray. Since there is no reasonable basis for neglecting the effects of the control rods, GPC has determined that by accurately accounting for the current plant design, a future change in design that reduces the available decontamination factor of the containment spray can be avoided.

2. The statistical convolution method described in WCAP-10125 for the evaluation of initial fuel rod to nozzle growth gap has not been approved and should not be used for VANTAGE 5. Please describe the method used to determine initial fuel rod to nozzle growth gaps in your evaluation of fuel rod performance.

   <u>Response</u>

   To determine the initial fuel rod to nozzle growth gap as a result of fuel rod irradiation effects, the worst case fabrication tolerances (mechanistic method) were evaluated in the fuel rod performance analysis (ELV-02166 Enclosure 4, page 8) rather than using the statistical convolution method described in WCAP-10125. This is in compliance with Condition 1 of the VANTAGE 5 NRC Safety Evaluation Report, WCAP-10444-P-A.

3. Is the assumption of loss of power during a steam line break the most conservative case? If so, was this assumed in the reference analysis?

Response

The VEGP-specific steam system piping failure event (FSAR subsection 15.1.5) was not affected by the transition to VANTAGE 5 fuel or the F-delta-h peaking factor limit increase as discussed in the GPC submittal (Enclosure 4, appendix A, section 15.1.5, page A-15.1-8). The VEGP radiological consequences evaluation is addressed in the GPC submittal (Enclosure 4, Appendix C, Section 1.0, page C-1).

The current VEGP FSAR specific analysis is performed with and without offsite power available. The case with offsite power available is the most limiting because the RCPs continue to operate, which produces a faster RCS cooldown rate.

4. Provide a more detailed justification for each Technical Specification change requested. Simply stating that the justification for the change is to provide operational flexibility is not sufficient.

Response

The response to this question has been made in a separate letter, ELV-02363 dated January 29, 1991. As a result of the meeting with the NRC staff on February 13, 1991, GPC has identified two Bases revisions that need additional discussion. That discussion is provided below.

Bases section 2.1.1 included a change that replaced the numerical values of the enthalpy hot channel factor and the power factor multiplier with the appropriate referenced parameter variables found in the Core Operating Limits Report (COLR). This change to the bases is consistent with the previously accepted changes in terminology associated with the COLR. The change to this Bases section does not indicate any change to the Bases for any specification. It only represents a change in the location of the values from the Bases to the COLR, which was previously approved by the NRC. Also, Bases sections 3/4.2.2 and 3/4.2.3 were revised to reference the Technical Specifications section from which figure 3.1-3 was relocated to the COLR.

The phrase "(break flow $\geq$ 3.0 $ft^2$)" was deleted from Bases section 3/4.5.4. The phrase as used in the Bases implies that a large break LOCA is defined as being greater than 3 $ft^2$. For the Vogtle Electric Generating Plant a large break LOCA is defined as larger than 1 $ft^2$. This is clearly stated in FSAR Section 15.6.5.1 and is not being changed as a result of the reanalyses associated with the use of VANTAGE 5 fuel. Since this value is clearly stated in FSAR paragraphs 3.9.N.2.5.2.2 and 15.6.5.1, GPC does not believe that it is necessary to restate it in this section of the Bases. The change to the Bases does not affect any existing Technical Specifications value nor does it affect any of the analyses or Technical Specifications changes requested in conjunction with the use of VANTAGE 5 fuel.

At the meeting on February 13, the NRC stated that it considered the deletion of the phrase "by RTD manifold instrumentation" from table 2.2-1 as a request to remove the RTD manifolds. Georgia Power Company has previously stated the RTD manifolds would not be removed without an appropriate submittal to the NRC. Georgia Power Company is currently reviewing the effects of RTD manifold deletion to determine the necessary contents of that submittal, which is scheduled to be made by the end of March 1991.

5. Is Vogtle only going to operate under relaxed axial offset control (RAOC) or will base load operation also be available? If so, TS 3.2.1 should so indicate.

Response

The proposed relaxed axial offset control (RAOC) specification in conjunction with the proposed heat flux hot channel factor ($F_Q$) specification provide sufficient operating margin and flexibility to envelope anticipated power shapes during either load follow or base load operation. Even though $F_Q$ margin can be gained by allowing either load follow or base load operation, GPC prefers only the load follow option which is more limiting in $F_Q$ operating space.

6. Current TS require an $F_Q$ measurement after exceeding by 10% the thermal power at which $F_Q$ was last measured . What is the justification for increasing this measurement requirement to 20% of rated thermal power in proposed Surveillance Requirement 4.2.2.2.e?

Response

Current Technical Specification 4.2.2.2.d.1 requires remeasuring $F_{xy}$ (which is used to verify compliance with $F_Q$ requirements) after a 20% power increase if the last computed $F_{xy}$ at less than full power conditions is greater than the $F_{xy}$ limit at RTP but less than the applicable limit for that lower power level. Therefore, the proposed Technical Specification is consistent with the current requirement except that $F_Q$ is measured directly and remeasured after a 20% power increase above the power level at which $F_Q$ was last measured regardless of the relationship of the measured FQ to the limiting value of $F_Q$.

Proposed Specification 4.2.2.2.e defines the remeasurement schedule for the heat flux hot channel factor ($F_Q$) surveillance. The remeasurement requirement 4.2.2.2.e.1 is not an assumption in the safety analyses; however, the remeasurement requirement is considered prudent to ensure that the $F_Q$ limit is not exceeded after raising the reactor power level from the power level at which $F_Q$ was last measured. The choice of 10% vs. 20% is arbitrary. The requirement to remeasure $F_Q$ following a 20% increase in power provides more operating flexibility without affecting plant safety or violating fuel design limits. The requirement to remeasure $F_Q$ after a 10% increase in power may result in taking additional flux maps that are unnecessary.

Therefore, a 20% power level increase is acceptable, is consistent with the current Technical Specification, and continues to meet the intent of the remeasurement schedule.

7. What is the justification for removing the discussion of available margin for offsetting fuel rod bow penalties from Bases 3/4.2.2 and 3/4.2.3?

Response

The discussion concerning the use of available DNBR margin to offset fuel rod bow penalties was relocated to Bases section 2.1.1 which is a more appropriate section for discussion of DNBR limits and DNB methodology. The 9.1% generic margin is no longer appropriate for the VEGP DNB analyses which use the WRB-1 and WRB-2 DNB correlations with the Revised Thermal Design Procedure. The VANTAGE 5 fuel is analyzed using the WRB-2 DNB correlation with design limit DNBR values of 1.24 and 1.23 for the typical cell and thimble cells, respectively. The LOPAR fuel is analyzed using the WRB-1 DNB correlation with design limit DNBR values of 1.23 and 1.22 for the typical cell and thimble cells, respectively. Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analysis limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility. This discussion is provided in the Bases section 2.1.1. See the response to Question 10 for further discussion of DNBR margins and penalties.

8. Does the criticality analyses of the new and spent fuel storage racks at Units 1 and 2 allow for the storage of VANTAGE 5 fuel assemblies?

Yes, the current criticality analyses for the new and spent fuel storage racks for both units allow for the storage of VANTAGE-5 fuel assemblies. These analyses were described in GPC letter ELV-00511 dated June 12, 1989. The criticality analyses were based on the Westinghouse Optimized Fuel Assembly (OFA). There are no differences in the design features of the OFA and VANTAGE-5 assemblies that would change the criticality analyses.

9. The reload report indicates an increase in the nuclear enthalpy rise hot channel factor (F-delta-h) limits for future cycles and also specifies different limits for VANTAGE 5 fuel (1.65) compared to LOPAR fuel (1.57). Explain why the F-delta-h TS 3/4.2.3 was not modified accordingly.

Response

Since the F-delta-h limits are specified in the VEGP Core Operating Limits Report (COLR) for each cycle of reactor operation, no changes to Technical Specifications section 3/4.2.3 were required. The F-delta-h limits of 1.65 and 1.57 for VANTAGE 5 and LOPAR fuel, respectively, will be included in the COLR for each applicable unit's cycle of operation.

10. What is the magnitude of the transition core penalty for VANTAGE 5 fuel? Justify that sufficient margin exists between the design and safety limit DNBR values to cover the rod bow and transition core penalties.

Response

As discussed in the GPC submittal (Enclosure 4, section 4.0, page 24), plant specific DNBR margin was maintained by performing the safety analyses to meet DNBR limits higher than the design limit DNBR values. A fraction of the available DNBR margin is utilized to accommodate the transition core penalty. For VANTAGE 5 fuel, this transition core penalty is a function of the number of VANTAGE 5 fuel assemblies in the core. This transition core penalty for VANTAGE 5 fuel is that given in the NRC-approved Westinghouse Topical Report, WCAP-11837-P-A, "Extension of Methodology for Calculating Transition Core DNBR Penalties," dated January 1990. There is no transition core penalty for [illegible] LOPAR fuel.

A summary of the design and safety analyses limit DNBR values is presented in the table below. In addition to the rod bow and transition core penalties there exists other DNBR penalties for RCS flow anomaly and Ventrak instrumentation bias and DNBR benefits associated with use of thimble plugs in reload core designs. Since the DNBR margin, penalties, and benefits can change for each reload design, the net DNBR margin is evaluated as part of the cycle reload design process. The following table is presented to show that margin exists for the first cycle transition to VANTAGE 5 fuel. Since individual DNBR penalties and benefits are proprietary to Westinghouse, these DNBR penalties and benefits are combined into an approximate value. The margin assessment below is based on loading 72 VANTAGE 5 fuel assemblies in the first transition cycle currently planned for Vogtle 1 Cycle 4.

DNBR MARGIN SUMMARY FOR VEGP UNITS 1 & 2

| | LOPAR | VANTAGE 5 (First Cycle) | VANTAGE 5 (Full Core) |
|---|---|---|---|
| Design Limit DNBR | | | |
| Typical Cell | 1.23 | 1.24 | 1.24 |
| Thimble Cell | 1.22 | 1.23 | 1.23 |
| Safety Analyses Limit DNBR | | | |
| Typical Cell | 1.35 | 1.51 | 1.51 |
| Thimble Cell | 1.34 | 1.49 | 1.49 |
| DNBR Margin | | | |
| Typical Cell | 8.8% | 17.8% | 17.8% |
| Thimble Cell | 8.9% | 17.4% | 17.4% |
| DNBR Resulting Penalty for Rod Bow, Transition Core, RCS Flow Anomaly, Instrumentation Bias, less benefit for Thimble Plug Use | < 4.0% | < 15.0% | < 5.0% |
| Net DNBR Margin | > 4.0% | > 2.0% | > 12.0% |

Based on the draft Merits Technical Specification work effort to date and the fact that VEGP specific net DNBR margin for each unit changes on a reload-to-reload basis, GPC has included only the design limit DNBR values in the Bases, since these values will not change as a result of a reload cycle design.

11. What is the decrease in RCS design flow from 382,800 gpm to 374,400 gpm attributed to?

<u>Response</u>

The reduced thermal design flow value of 374,400 gpm is based on a future accommodation of up to 10% steam generator tube plugging. The current VEGP thermal design flow value of 382,800 gpm accommodates only a small fraction of steam generator tube plugging (< 1%). The reduced thermal design flow and the 10% steam generator tube plugging were assumed in the DNBR analyses for VANTAGE 5. However, approval for reducing the minimum measured flow (TS 3/4.2.5) to take advantage of the reduced thermal design flow and allowance for 10% steam generator tube plugging, is not being sought at this time.

Assuming reduced flow and increased plugging in the VANTAGE 5 DNBR analyses provides conservative results and eliminates the need for reanalysis of DNB transients in order to request implementation of reduced flow and increased tube plugging. Full implementation of these changes requires additional work that is outside the scope of the GPC VANTAGE 5 submittal, therefore it is not being requested as part of the VANTAGE 5 submittal.

12. Define minimum measured flow (MMF) and thermal design flow (TDF) and show how they were determined.

Response

Thermal design flow is the value of flow assumed in the safety analyses. The minimum measured flow (the value specified in the LCO of T.S. 3/4.2.5) is the thermal design flow increased by the RCS flow measurement uncertainty. Satisfying the minimum measured flow requirement ensures that the core flow will be greater than or equal to the thermal design flow assumed in the safety analyses.

13. Specify which safety analyses, if any, are performed by GPC.

Response

All VANTAGE 5 safety analyses provided in the GPC submittal were performed by Westinghouse. None were performed by GPC.

14. The transient analyses assume a total negative reactivity insertion following a trip of 4.0% delta k/k. Verify that this amount of RCCA reactivity is available for all operating conditions during the cycle.

Response

The total negative trip reactivity insertion input parameter of 4.0% delta k/k is confirmed for each cycle as part of the cycle reload design process in accordance with the Topical Report WCAP-9272-P-A, "Westinghouse Reload Safety Methodology," dated July 1985. The 4.0% delta k/k assumption is a generic value used in both the current Vogtle FSAR safety analyses and the VANTAGE 5 safety analyses. The minimum total negative trip reactivity insertion value of 4.0% delta k/k is valid for those safety analyses performed at initial full power operating conditions at the most limiting time in cycle life (i.e., beginning or end of cycle life). This assumption is conservative with respect to the trip reactivity worth calculated during each reload design process. The total trip reactivity insertion is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position.

15. The RTDP for calculating DNB limits has been approved with certain conditions imposed on its implementation because of the sensitivity of the method to changes in correlations and codes used. Explain how each of these conditions are accounted for in the use of the RTDP for VEGP Units 1 and 2.

Response

The NRC staff position identified seven conditions o the implementation of the RTDP procedure for calculating DNB limits. The analyses presented with our letter ELV-02166 dated November 29, 1990, were performed within the limits of those conditions. Each of the conditions is listed and discussed below.

A. Sensitivity factors used for a particular plant and their ranges of applicability should be included in the Safety Analysis Report or reload submittal.

   Sensitivity factors were evaluated for the DNB correlations, THINC-IV model and parameter values for the specific application of RTDP to VEGP. The factors and their range of application will be supplied to the NRC with proper notification of proprietary information under a separate cover letter.

B. Any change in DNB correlation, THINC-IV correlations, or parameter values listed in Table 3-1 of WCAP-11397-P-A outside of previously demonstrated acceptable ranges require re-evaluation of the sensitivity factors and of the use of Equation (2-3) of the topical report.

   See response to item A above.

C. If the sensitivity factors are changed as a result of correlation changes or changes in the application or use of the THINC code, then the use of an uncertainty allowance for application of Equation (2-3) must be reevaluated, and the linearity assumption made to obtain Equation (2-17) of the topical report must be validated.

   Equation (2-3) and the linearity approximation made to obtain Equation (2-17) are still valid for the VEGP application. The sensitivity factors, operating parameters, and THINC-IV model used in this application are consistent with those used in WCAP-11397-P-A.

D. Variances and distributions for input parameters must be justified on a plant-by-plant basis until generic approval is obtained.

   The plant-specific variances and distributions for this application are provided in Proprietary Reports WCAP-12460 (for RTD bypass Loop) and WCAP-12462 (for RTD bypass loop elimination). The nonproprietary versions of these reports are WCAP-12461 and 12463, respectively. All four WCAPs were supplied to the NRC under separate cover letter.

E. Nominal initial condition assumptions apply only to DNBR analyses using RTDP. Other analyses, such as overpressure calculations, require the appropriate conservative initial condition assumptions.

Nominal initial conditions were applied only to DNBR analyses which used RTDP.

F. Nominal conditions chosen for use in analyses should bound all permitted methods of plant operation.

Bounding nominal conditions were used in the DNBR analyses which used RTDP.

G. The code uncertainties specified in table 3-1 (± 4 percent for THINC-IV and ± 1 percent for transients) must be included in the DNBR analyses using RTDP.

The code uncertainties specified in table 3-1 of WCAP-11397-P-A were included in the DNBR analyses using RTDP.

16. Why is the STDP used for the uncontrolled RCCA bank withdrawal event from a shutdown or low power condition whereas the RTDP is used for the event initiated from power?

Response

The RTDP methodology is used for only those transients which have DNB as a limiting criterion and are initiated at or near full power conditions. For those transients not having a DNB criterion or initiating at zero or low power conditions, the STDP methodology is used. Therefore, STDP was used for the uncontrolled RCCA bank withdrawal event initiated from a shutdown or low power condition, and RTDP was used for the same event initiated from full power conditions.