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U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555

VOGTLE ELECTRIC GENERATING PLANT
NRC DOCKETS 50-424, 50-425
OPERATING LICENSES NPF-68, NPF-81
REVISION TO TECHNICAL SPECIFICATIONS
CONTROL ROD MATERIAL AND ROD INSERTION LIMITS

Gentlemen:

In accordance with provision of 10 CFR 50.90 and 10 CFR 50.59, Georgia Power Company (GPC) hereby proposes to amend the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 Technical Specifications, Appendix A to Operating Licenses NPF-68 and NPF-81.

The proposed changes to the Technical Specifications revise the rod insertion limits of figure 3.1-3 to show the fully withdrawn position as 222 steps instead of 228 steps. The purpose of the change in the insertion limit is to allow a periodic change, within the range of 222 to 231 steps withdrawn, of the parked full out position of the control rods which will minimize the effects of control rod wear caused by fretting against upper internal control rod guide surfaces. Specification 3.1.3.4 will also be revised to assure that control rod drop time measurements are made from the physical fully withdrawn position (231 steps withdrawn). In addition, a revision to Technical Specification 5.3.2 is being proposed that will indicate that control rods may utilize an absorber material of either hafnium or silver-indium-cadmium. The use of either type of absorber material was described in the VEGP Final Safety Analysis Report (FSAR) and in the NRC's Safety Evaluation Report for VEGP. However, Section 5.3.2 of the Technical Specifications only described hafnium since that was the only absorber material in use at the time that the VEGP Units 1 and 2 licenses were issued. Current plans are to replace control rods with absorber material of hafnium with control rods that use silver-indium-cadmium, beginning at the next refueling outage which is scheduled for VEGP Unit 1 in the Spring of 1990.

GPC plans to begin the use of silver-indium-cadmium absorber material for the control rods and the revised rod insertion limits, starting with Cycle 3 of Unit 1 (Spring of 1990). These changes have been evaluated for Cycle 1 of Unit 2 and will be included in the reload evaluation for future cycles of both units. Therefore, GPC requests that the proposed Technical Specification revisions be applicable to both Units 1 and 2 at the beginning of Cycle 3 for VEGP Unit 1.

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Georgia Power



Enclosure 1 provides a detailed description of the proposed changes and the circumstances necessitating the change request.

Enclosure 2 provides the bases for a determination that the proposed changes do not involve significant hazards considerations.

Enclosure 3 provides instructions for incorporating the proposed changes into the Technical Specifications. The proposed revised pages for the combined VEGP Units 1 and 2 Technical Specifications are provided as Enclosure 3.

In accordance with 10 CFR 50.91, the designated state official will be sent a copy of this letter and all enclosures.

Mr. W. G. Hairston, III states that he is a Senior Vice President of Georgia Power Company and is authorized to execute this oath on behalf of Georgia Power Company and that, to the best of his knowledge and belief, the facts set forth in this letter and enclosures are true.

GEORGIA POWER COMPANY

By: W. G. Hairston III
W. G. Hairston, III

Sworn to and subscribed before me this 24th day of August, 1989.

Sherry Ann Mitchell
Notary Public
MY COMMISSION EXPIRES DEC. 15, 1992

WGH,III/HWM/gm

Enclosures

xc (see next page)

Georgia Power



xc w/enclosures::

Georgia Power Company
Mr. C. K. McCoy
Mr. G. Bockhold, Jr.
Mr. R. M. Odom
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U. S. Nuclear Regulatory Commission
Mr. S. D. Ebneter, Regional Administrator
Mr. J. B. Hopkins, Licensing Project Manager, NRR
Mr. J. F. Rogge, Senior Resident Inspector, Vogtle

ENCLOSURE 1

VOGTLE ELECTRIC GENERATING PLANT
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REVISION TO TECHNICAL SPECIFICATIONS
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## BASIS FOR PROPOSED CHANGE

### Proposed Changes:

The following changes are included in this submittal:

1. In Specification 3.1.3.4 for control rod drop time the word "physical" is inserted into the Limiting Condition for Operation (LCO) such that it states "The individual shutdown and control rod drop time from the physical fully withdrawn position..."

2. In Specification 3.1.3.5 for shutdown rod insertion limits the LCO is restated as " All shutdown rods shall be withdrawn to a position greater than or equal to 222 steps" instead of "All shutdown rods shall be fully withdrawn".

   In the ACTION statement the phrase "not fully withdrawn" is replaced by "inserted to a position less than 222 steps," and the phrase "Fully withdraw the rod, or" is replaced by "withdraw the rod to a position greater than or equal to 222 steps, or"

3. In Specification 4.1.3.5, the Surveillance Requirement for Shutdown Rod Insertion Limit, the phrase "Each shutdown rod shall be determined to be fully withdrawn:" is replaced by "Each shutdown rod shall be determined to be withdrawn to a position greater than or equal to 222 steps:"

4. Figure 3.1-3 "Rod Bank Insertion Limits Versus Thermal Power" is replaced with a Figure that is identical except that it shows the fully withdrawn position as 222 instead of 228.

5. In Surveillance Requirement 4.10.1.1 the phrase "each control rod either partially or fully withdrawn" is changed to "each control rod not fully inserted".

6. In the bases for Specifications 3/4.2.2 and 3/4.2.3, part "b" is replaced with "Control rod banks are sequenced with a constant tip-to-tip distance between banks as defined by Figure 3.1-3."

7. In Specification 5.3.2 the description of the absorber composition is revised by inserting the phrase "and/or 80% silver, 15% indium, and 5% cadmium".

### Basis

Recent operating experience at other nuclear plants has identified control rod wear as a potential problem. This wear is a result of the fretting of control

BASIS FOR PROPOSED CHANGE

rods against upper internal control rod guide surfaces. Because the control rods operate for extensive periods of time in the withdrawn position, the point of wear is local and is determined by the withdrawn parked position. By allowing for a range of parked positions, the potential effects of fretting induced wear can be reduced. The current control rod insertion limits require that fully withdrawn rods be at least 228 steps withdrawn. The physical limit to rod withdrawal is 231 steps. This proposed change reduces the control rod insertion limit to 222 steps withdrawn. This change will revise the all rods out (ARO) parked position such that the control rods can be periodically repositioned between 222 and 231 steps withdrawn. Prior to reducing the rod insertion limits it is necessary to review the potential effects on the LOCA and Non-LOCA Accidents to assure that the analyses of these accidents will remain valid for the new limits. This has been done and the results are discussed in Enclosure 2. Since this change to the Technical Specifications would allow the control rods to be placed at a variety of positions between 222 and 231 steps withdrawn, the Technical Specifications were reviewed to assure that the term "fully withdrawn" as used for control rods would not be confusing. As a result of this review the phrase "physical fully withdrawn" was chosen for the control rod drop time specification in order to assure that control rod drop times are measured from the most withdrawn position possible. In specification 4.10.1.1 the wording was revised to eliminate the use of the phrase "fully withdrawn" without changing the requirements of the specification.

An additional change has been included with this Technical Specification change request that concerns control rods but is unrelated to control rod position. This change is to that portion of Section 5.3.2 of the Technical Specifications that describes the neutron absorber material contained in the control rods. The design of VEGP Units 1 and 2 as described in the FSAR allows for the use of either hafnium or silver-indium-cadmium absorber material. This aspect of the design was also noted in the NRC staff's Safety Evaluation Report for VEGP Units 1 and 2. However, since only hafnium was used in the initial core design of VEGP Units 1 and 2, it was the only absorber material mentioned in that section of the Technical Specifications. GPC expects to replace the current hafnium control rods with rods that use silver-indium-cadmium absorber material at the next refueling outage for VEGP Unit 1 which is scheduled for the Spring of 1990. Similar absorber material will be used in replacement control rods for VEGP Unit 2 at its first refueling outage. Therefore, this change which is being requested for the Technical Specifications is consistent with the current design presented in the FSAR and as previously evaluated by the NRC staff. The use of this absorber material was also reviewed by Westinghouse to determine that the previous considerations of the FSAR remain valid. The results of this evaluation are also documented in Enclosure 2.

ENCLOSURE 2

VOGTLE ELECTRIC GENERATING PLANT
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REVISION TO TECHNICAL SPECIFICATION
CONTROL ROD MATERIAL AND ROD INSERTION LIMITS

## 10 CFR 50.92 EVALUATION

Pursuant to 10 CFR 50.92, Georgia Power Company (GPC) has evaluated the attached proposed amendment to the VEGP Units 1 and 2 Technical Specifications and has determined that operation of the facility in accordance with the proposed amendment would not involve significant hazards considerations. The basis for this determination follows:

### Background

The current control rod insertion limits as defined in Technical Specification Figure 3.1-3 identifies the maximum RCCA insertion limit for control banks A, B and C at 100% power as 228 steps withdrawn. The Figure also implies a constant tip-to-tip distance between control banks. The maximum number of steps that a RCCA can physically be withdrawn is 231 steps. During extended periods of operation the control rods may be parked in an all rods out (ARO) position of between 228 and 231 steps. The practice at VEGP is currently to park the control rods at 228 steps withdrawn. Past operational history at other plants has indicated that operation with control rods withdrawn to the same position for long periods of time can result in excessive localized control rod wear due to fretting against the upper internals guide surface as a result of flow induced vibration. In order to minimize the potential effects of such wear, GPC proposes to reposition control rods within a range from 222 to 231 steps withdrawn. This requires a change to the RCCA insertion limits, which is included in this proposed change to the Technical Specifications. Even though the amount of overlap between control banks will vary with the ARO position, the tip-to-tip distance between banks will not change as a result of repositioning.

In addition, GPC has decided to replace the control rods that are currently used in VEGP Units 1 and 2 with control rods of the same design but utilizing silver-indium-cadmium as the absorber material instead of hafnium. The use of either type of control rod absorber material has previously been reviewed by the NRC for use at VEGP but it was not included in the Technical Specifications. Therefore, this proposed change to the Technical Specifications also includes a revision to the Design Features section to indicate the use of either hafnium (Hf) or silver-indium-cadmium (Ag-In-Cd) as an absorber material in the control rods.

The effects of these revisions to the control rod insertion limits and the use of silver-indium-cadmium absorber material on the accidents that have been analyzed for VEGP Units 1 and 2 have been evaluated. The results of these evaluations are presented in the analysis that follows.

ENCLOSURE 2 (CONTINUED)
CONTROL ROD MATERIAL AND ROD INSERTION LIMITS

## 10 CFR 50.92 EVALUATION

### Analysis

The current Technical Specifications allow individual control rods to be parked at any ARO position of 228 to 231 steps withdrawn. The proposed revision would change this to a range of 222 to 231 steps withdrawn and allow the use of Ag-In-Cd control rods. In order to evaluate the effects of these Technical Specification changes, the effects of the proposed changes on the potentially affected safety analyses were evaluated and are discussed below.

### 1.0 Non-LOCA Accident Analyses

The RCCA's are modeled in the non-LOCA analyses in terms of rod worth, total trip reactivity, drop time and reactivity insertion versus time. It has been determined, via a generic evaluation, that Ag-In-Cd rods used in place of Hf rods will not cause any changes to the safety analysis input parameters used for Vogtle 1 and 2 and that such parameters are appropriate for either rod material. The generic evaluation consisted of performing two parallel core designs, varying only the control rod material. Results confirmed that parameters such as control rod worths and peaking factors remained essentially the same. This conclusion also applies to any combination of hafnium and Ag-In-Cd rods in the core. The two different RCCA types are neutronically equivalent. It is therefore concluded that the assumptions to the non-LOCA analyses do not change dependent upon the control rod material and there is no affect on these analyses as a result of this change. The non-LOCA results presented in the FSAR remain bounding.

The data used in the non-LOCA safety analyses was also reviewed with respect to the change in rod insertion limits. The input parameters to the safety analyses remain unchanged and are not affected by the proposed parked position range between 222 and 231 steps withdrawn.

Specifically, the RCCA worth, total trip reactivity, shutdown margin, RCCA drop time and reactivity versus time curve remain valid. In addition, there are no changes to the physics parameters assumed in the safety analyses. These results apply to both Hf and Ag-In-Cd control rods. Therefore the safety analyses results, as presented in the FSAR remain bounding.

The rod insertion limits Technical Specification change will be implemented during the Unit 1 Cycle 2 to 3 outage (Spring 1990). Thus, the revised Radial Peaking Factor Limit Report for Cycle 3 will incorporate the effects of the revised rod insertion limits. Revised radial peaking factors for the remainder of Unit 2 Cycle 1 will be identified via a mid-cycle revision to the Radial Peaking Factor Limit Report in accordance with Technical Specification 6.8.1.6.

### 2.0 LOCA and LOCA-Related Analyses

#### Small Break LOCA

The small break LOCA analysis for the Vogtle Units is described in FSAR Section 15.6.5. The current FSAR small break LOCA analysis was performed

with the WFLASH and LOCTA Evaluation Model. This Evaluation Model assumes the reactor core is brought to a subcritical condition by the trip reactivity of control rods. Since the rod drop time and trip reactivity requirements assumed in the safety analyses remain valid for this change in rod material and for the change in the rod insertion limits, there will be no effect on the current FSAR small break LOCA analysis.

Large Break LOCA

The large break LOCA analysis for the Vogtle Units is described in FSAR Section 15.6.5. The current FSAR large break LOCA analysis was performed with the 1981 Evaluation Model. This large break LOCA Evaluation Model does not take credit for the negative reactivity introduced by the control rods. During a large break LOCA, the reactor is brought to a subcritical condition by the presence of voids in the core. Since credit was not taken for the control rods there will be no effect on the current FSAR large break LOCA analysis.

Post-LOCA Long-Term Core Cooling

The Westinghouse Evaluation Model commitment is that the reactor will remain shutdown by borated ECCS water from the RCS/sump following a LOCA. Since credit for the control rods is not taken for a large break LOCA, the borated ECCS water provided by the accumulators and the RWST must have a boron concentration, when mixed with other water sources, which will result in the reactor core remaining subcritical, assuming all control rods out. Since there is no credit taken for control rods, there will be no change in the calculated RCS/sump boron concentration after a postulated LOCA, and there is no adverse effect on the post-LOCA long-term core cooling requirement as presented in the FSAR. It is conservative to assume no credit for control rod insertion since this results in a more reactive condition upon which the evaluation is performed.

Hot-Leg Switchover to Prevent Boron Precipitation

Discussion of the hot leg switchover time is found in the Vogtle FSAR Section 6.3.2.8, Section 6.3.3.3, and Section 15.6.5.2. The hot leg switchover time is determined for inclusion in emergency procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This time is dependent on power level, and the RCS, RWST, and accumulator water volumes and boron concentrations. Changing the insertion limit or changing the control rod material does not affect either the power level, or the maximum boron concentrations assumed for the RCS, RWST and accumulators in the hot leg switchover calculation. Therefore, the post-LOCA hot leg switchover time presented in the FSAR remains valid.

## 10 CFR 50.92 EVALUATION

### Rod Ejection Mass Releases

The rod ejection mass releases for dose calculations for the Vogtle Units is presented in FSAR Table 15.4.8-2. The current FSAR rod ejection mass release calculation for the Vogtle Units was performed with the WFLASH Evaluation Model. Since the rod drop time does not change, the change in the control rod material and rod insertion limits does not create an adverse effect on the rod ejection mass releases as presented in the FSAR.

### LOCA Hydraulic Forcing Functions

The blowdown hydraulic forces resulting from a LOCA are considered in Section 3.9(N).1.4.6 of the Vogtle FSAR. Based on the current LOCA analyses, the shortest time calculated for tripping the control rods is about 0.5 seconds after a postulated LOCA. The peak loads generated on the reactor vessel and internals as result of a postulated LOCA occur typically between 10 to 50 milliseconds (.01 to .05 seconds) after the break and these loads have subsided well before 500 milliseconds (.50 seconds). Since the hydraulic forces as result of a LOCA have peaked and subsided before the time the control rods are calculated to trip, the change of the control rod material and change in rod insertion limits will have no effect on the structural analysis presented in the Vogtle FSAR.

### 3.0 Containment Integrity

The change in control rod material and rod insertion limits has no effect on the mass and energy release assumptions used in the containment integrity analyses. Therefore, the results presented in the FSAR remain bounding for these changes.

### 4.0 Steam Generator Tube Rupture

During a steam generator tube rupture, the loss of coolant due to the primary to secondary break flow results in a decrease in the RCS pressure and reactor trip occurs due to low pressurizer pressure. The control rod scram was modeled utilizing the total trip reactivity and the normalized reactivity versus time after trip data. Since neither of these are affected, the steam generator tube rupture analysis assumptions and results remain bounding for the change in control rod material as well as for the change to rod insertion limits.

### 5.0 Mechanical Performance Evaluation

The RCCA's are designed to fall via gravity into the reactor core when power is removed from the drive mechanisms. The time to fall into the dashpot region

of the core is currently assumed to be 2.2 seconds. It has been determined that this rod drop time assumption is valid for either rod material such that there will be no change to the value provided for use in the accident analyses.

Rod repositioning is being implemented as a management scheme to mitigate the effects of wear and prolong the life of the control rods to perform their functional requirements. These functional requirements include providing negative reactivity, maintaining geometric configuration for expected duty and maintaining continuous free movement. Mechanically, the repositioning will not affect the operation of the control rod drive mechanisms and as such does not represent a design or operational issue. Therefore, the conclusions of any analyses which assume a 2.2 second rod drop time remain valid for either rod material and for the range of parked positions from 222 to 231 steps.

RESULTS

The safety analyses have been determined to be unaffected by the changes in control rod material or rod insertion limit. Also, there are no adverse effects to any safety related fluid system performance, radiological consequences or equipment qualification issues as a result of these modifications. Based on this and the additional information presented above. The following has been concluded:

1. The change in rod insertion limits and use of either rod material will not increase the probability of an accident previously evaluated in the FSAR. The evaluations presented above have determined that the rod drop time and other associated parameters assumed in the safety analyses have not changed. Additionally, there are no new failure modes introduced that would increase the probability of any reactivity anomaly occurring. Therefore, all current safety analyses will remain valid with these changes.

2. The change in rod insertion limits and use of either rod material will not increase the consequences of an accident previously evaluated in the FSAR. It has been determined that all safety related design criteria have been met assuming these changes. These changes will have minimal impact on core behavior and will not require any setpoint changes nor will the operability of the units be affected. Future reload safety evaluations will review the ARO position to assure continued conformance with the design parameters.

3. The change in rod insertion limits and use of either rod material will not create the possibility of a new or different kind of accident from any accident previously evaluated in the FSAR. No new failure mechanisms or new limiting single failures have been identified as a result of these changes.

## 10 CFR 50.92 EVALUATION

4. The change in rod insertion limits and the use of either rod material will not decrease the margin of safety in the bases to any Technical Specification. Since it has been determined that all accident analysis acceptance criteria and design and performance requirements continue to be met, there is no reduction to any previously defined margin of safety in the Technical Specifications.

It is expected that these revisions to the Technical Specifications will become effective at the beginning of Cycle 3 for VEGP Unit 1. In addition, since Unit 2 Cycle 1 will still be in progress when these revised Technical Specifications become effective, the changes will also be acceptable for immediate implementation on Unit 2 Cycle 1.

### Conclusion

Based on the preceding analysis, GPC has determined that the proposed change to the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated or involve a significant reduction in a margin of safety. Therefore, GPC concludes that the proposed changes meet the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.