Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-3195

Mailing Address
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 868-5581

the southern electric system

W. G. Hairston, III
Senior Vice President
Nuclear Operations

January 29, 1991

ELV-02363
0759

Docket Nos. 50-424
50-425

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

Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
ADDITIONAL INFORMATION FOR VANTAGE-5 FUEL
RELATED TECHNICAL SPECIFICATION CHANGES

Georgia Power Company (GPC) letter ELV-02166 dated November 29, 1990 transmitted a request for Technical Specification changes associated with the use of VANTAGE-5 fuel. At the request of the NRC we are providing additional information in the attachment to this letter. The initial submittal was lengthy and requested a number of different Technical Specification changes. The supplemental information is intended to clarify the evaluation of each group of Technical Specification changes relative to 10 CFR 50.92 . This information does not change any of the previously submitted information nor does it alter any of the conclusions presented in our letter of November 29, 1990. In order to support the review schedule, GPC is prepared to meet with the NRC staff to discuss the various aspects of this application and how it relates to GPC's long term fuel management strategy.

GPC is in the process of evaluating the effects of removing the Resistance Temperature Detector (RTD) bypass manifolds. We expect to complete this evaluation in the Spring of 1991. At this time, GPC does not expect that the removal of the RTD bypass manifolds will require further Technical Specification changes because margin has been incorporated into the analyses performed for the VANTAGE-5 transition to account for the potential effects of their removal. However, our letter ELV-02166 is not intended to be a request for RTD bypass manifold removal. Should GPC decide to proceed with RTD bypass manifold removal, an appropriate submittal will be made to the NRC. GPC is prepared to discuss this, as well as any other topics related to our previous submittal at the meeting proposed above.

Sincerely,

W. G. Hairston

W. G. Hairston, III

WGH,III/HWM/gm
Enclosure

A001
1/1

Georgia Power

xc: Georgia Power Company
Mr. C. K. McCoy
Mr. W. B. Shipman
Mr. P. D. Rushton
Mr. R. M. Odom
NORMS

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

VOGTLE ELECTRIC GENERATING PLANT
SUPPLEMENTAL INFORMATION
NO SIGNIFICANT HAZARDS ANALYSIS FOR
VANTAGE-5 FUEL TECHNICAL SPECIFICATION CHANGES

Georgia Power Company (GPC) letter ELV-02166 dated November 29, 1990 requested Technical Specifications changes associated with analyses performed in support of the planned transition to VANTAGE-5 fuel. Letter ELV-02166 included an evaluation to demonstrate that the proposed changes to the Technical Specifications did not involve any significant hazards considerations as defined by 10 CFR 50.92. The proposed Technical Specifications revisions can be grouped into six different changes. At the NRC's request, GPC has prepared this supplement to the previously submitted information. This supplement is intended to specifically address the question of significant hazards considerations for each of the Technical Specification changes. Where appropriate, the following supplemental evaluations refer to the information contained in the Enclosures and Appendices of GPC letter ELV-02166.

MINIMUM RWST SOLUTION TEMPERATURE

The proposed changes to Technical Specifications 3/4.1.2.5, 3/4.1.2.6, and 3/4.5.4 will reduce the RWST minimum solution temperature for the limiting condition for operation (LCO) from 54 °F to 44 °F and the associated surveillance limit from 50 °F to 40 °F. The LCO limit of 54 °F includes a 4°F uncertainty in measurement of the RWST solution temperature. The 40°F surveillance requirement establishes a minimum outside air temperature at which the RWST LCO solution temperature must be verified at least once per 24 hours.

The proposed Technical Specification changes identified above will provide additional operating flexibility by increasing the range of temperature within which the RWST will be available as an OPERABLE borated water source. Safety analyses which are sensitive to minimum RWST solution temperature have been reanalyzed as part of the VANTAGE-5 fuel transition report to confirm the acceptability of the temperature reduction. These analyses include Inadvertent Operation of the ECCS During Power Operation (FSAR 15.5.1), Small Break LOCA (FSAR 15.6.5) and Steam Generator Tube Failure (FSAR 15.6.3). The small break LOCA was reanalyzed using the NRC approved NOTRUMP methodology (WCAP-10080-A and WCAP-10081-A). In addition, the solubility of the RWST solution at the reduced temperature for the 2400-2600 ppm range of boron required per Technical Specifications 3.1.2.5 (b) (2), 3.1.2.6 (b) (2) and 3.5.4 (b) has also been confirmed as part of the VANTAGE-5 program.

Based on the information presented above and the analyses presented in the VANTAGE-5 fuel transition submittal (Enclosure 4 and Appendices A and B of letter ELV-02166), the following conclusions can be reached with respect to 10 CFR 50.92.

1. The reduced RWST minimum solution temperature does not increase the probability or consequences of an accident previously evaluated in the

FSAR. The RWST solution temperature is a parameter used in the analysis of the Steam Generator Tube Failure and Small Break LOCA accidents. Since RWST solution temperature is only used in the analysis of these events, it does not contribute as an initiator or affect the probability of occurrence. The Inadvertent Operation of the ECCS During Power Operation accident involves injection of borated RWST water into the RCS. Although RWST injection is part of the event, the initiator of the event is either operator error or a false electrical actuation signal, which are unaffected by RWST solution temperature. Therefore, a change in the RWST minimum solution temperature will not increase the probability of occurrence of this event.

The consequences of an accident previously evaluated in the FSAR are not increased due to the reduced RWST minimum solution temperature. Small Break LOCA and Inadvertent Operation of the ECCS During Power Operation are not evaluated for radiological consequences since they are not limiting transients with respect to prediction of offsite doses. A revised analysis has been performed for the Steam Generator Tube Failure event as part of the VANTAGE-5 submittal which documents that all doses are within the Standard Review Plan acceptance criteria. Therefore, the consequences to the public resulting from any accident previously evaluated in the FSAR have not significantly increased.

2. The reduced RWST minimum solution temperature does not create the possibility of a new or different kind of accident than those already evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures associated with the RWST are introduced as a result of the reduced allowable solution temperature. This reduced temperature condition in the RWST has no adverse effect and does not challenge the performance or integrity of any other safety related system. Therefore, the possibility of a new or different kind of accident is not created.

3. The margin of safety provided by the Technical Specifications relative to the RWST as a borated water source ensures that the reactor will remain subcritical under post-accident conditions. This inherently assumes that the defined boron concentration range will remain soluble at the RWST minimum solution temperature. By confirming that solubility at the reduced temperature is maintained, it is concluded that the operating envelope defined by the Technical Specifications continues to be bounded by the revised analytical basis. Therefore, the margin of safety provided by the RWST as a source of borated water is maintained and not reduced.

Based upon the preceding information, it has been determined that the proposed change to the Technical Specifications does not involve significant hazards considerations as defined by 10 CFR 50.92 (c).

INCREASE IN SHUTDOWN AND CONTROL ROD DROP TIME

Technical Specification 3/4.1.3.4 specifies the allowable shutdown and control rod drop time. This time is being changed from 2.2 seconds to 2.7

seconds. The increase is to account for a slightly higher pressure drop across the VANTAGE-5 fuel assembly due to the Intermediate Flow Mixer (IFM) grids and for the slightly smaller guide thimble diameter. The revised rod drop time was used for accident and transient reanalyses and evaluations presented in Enclosure 4. The results of these analyses and evaluations are described in Appendices A and B of Enclosure 4. The safety criteria and previously defined acceptance limits continue to be met. The 0.5 second increase in rod drop time allows for the slight increase in rod drop time expected from VANTAGE-5 fuel. The required verification of rod drop time remains the same. The revised Technical Specification limit is consistent with the value used for the accident and transient analyses described in Enclosure 4. Based on the results of those analyses the following conclusions can be reached regarding 10 CFR 50.92.

1. The increase in rod drop time will not result in an increase in the probability or consequences of any accident previously evaluated in the FSAR since the same surveillance requirements will be used to detect inoperable rods. The consequences of increased rod drop time have been evaluated and analyzed as reported in Enclosure 4 and determined to be within the acceptance limits.

2. The possibility of a new or different type of accident is not involved because the increase in rod drop time used in the analyses and in the Technical Specification is consistent with the design of the VANTAGE-5 fuel and does not indicate any new or different failure mechanism. Therefore, it does not indicate the possibility of a new or different type of accident.

3. The effects of the increased rod drop time have been included in the analyses and evaluations of accidents and transients included in Enclosure 4. These analyses demonstrated that the plant will remain within previously accepted limits, therefore the increase in the allowable rod drop time does not result in a significant reduction in a margin of safety.

Based upon the preceding information, it has been determined that the revision to the allowable rod drop time does not involve significant hazards considerations as defined in 10 CFR 50.92 (c).

AXIAL FLUX DIFFERENCE AND PEAKING FACTOR SURVEILLANCE

The revised analyses for the VANTAGE-5 fuel assumed the Relaxed Axial Offset Control (RAOC) methodology. This methodology has been previously approved by the NRC in WCAP-10216-P-A. The use of this methodology requires the replacement of Technical Specification 3/4.2.1 for Axial Flux Difference with the Technical Specification for RAOC. This specification is consistent with those previously approved by the NRC for other plants using RAOC.

The use of RAOC allows direct surveillance of the Heat Flux Hot Channel Factor (F_Q). Therefore, the surveillance requirements for Specification 3/4.2.2 are also being replaced with the appropriate F_Q surveillance requirements consistent with the surveillance requirements approved by the NRC for other plants using F_Q surveillance. In Technical Specification 3.2.2 ACTION a. the phrase "Overpower Δ T Trip Setpoints have been reduced at least 1%" is being changed to "Overpower ΔT Trip Setpoints (Value of K_4) have been reduced at least 1% (in Δ T span)". The addition of the two parenthetical phrases does not change the ACTION requirement and is only intended as clarification of the ACTION statement.

Section 6.8.1.6 of the Technical Specifications identifies analytical methods, previously approved by the NRC, that must be used to determine core operating limits. The use of the RAOC methodology requires that this section of the Technical Specification be revised to include WCAP-10216-P-A and to delete the references that were previously used for axial offset control.

Accidents and transients have been reanalyzed or evaluated for the use of VANTAGE-5 fuel using RAOC methodology and a higher F_Q peaking factor which will be specified in the Core Operating Limits Report. The small break LOCA and large break LOCA accidents were reanalyzed with the NRC approved codes NOTRUMP and BART/BASH(WCAP-9200-A, and WCAP-11524-A). The results of the non-LOCA and LOCA analyses are presented in Enclosure 4 and Appendices A and B. These analyses demonstrated that the NRC acceptance limits will continue to be met. Based on the results of those analyses the following conclusions can be reached regarding 10 CFR 50.92.

1. The use of the RAOC and F_Q surveillance does not increase the probability or consequences of an accident previously evaluated in the FSAR. The Technical Specification changes do not result in any physical changes in the plant or any other changes that could initiate an accident. The accidents previously evaluated in the FSAR have been reevaluated and the results indicate that the consequences have not significantly increased. The results of these analyses are presented in Enclosure 4 and Appendices A and B.

2. The use of RAOC and F_Q surveillance does not introduce the possibility of a new or different kind of accident than any previously evaluated in the FSAR. The operating limitations remain consistent with the analyses. The changes in the Technical Specifications do not result in the introduction of a new accident scenario, failure mechanism or limiting single failure.

3. The margin of safety provided by the Technical Specifications will not change significantly due to the use of the RAOC methodology and F_Q surveillance. This has been demonstrated by the reanalysis of transients and accidents presented in Enclosure 4 and Appendices A and B. The use of F_Q surveillance instead of F_{xy} provides a more direct method of demonstrating compliance with the Limiting Condition for Operation. The combination of RAOC and F_Q surveillance will continue to demonstrate that operation will remain within the constraints of the axial flux difference limits and will not result in total peaking factors that exceed the limit.

Based upon the preceding information, it has been determined that the use of RAOC and F_Q surveillances does not involve any significant hazards considerations as defined by 10 CFR 50.92 (c).

REACTOR CORE SAFETY LIMITS, REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS, AND DNB PARAMETERS

The Intermediate Flow Mixer (IFM) Grids VANTAGE-5 fuel design feature, the improved THINC-IV thermal-hydraulic design modeling methodology, the Revised Thermal Design Procedure (RTDP), and the WRB-1 and WRB-2 DNB correlations change the basis for determining DNBRs. This becomes the basis for proposed changes to the following Technical Specifications:

A. Technical Specification Figure 2.1-1 - The revised DNB methods allow for revision of the Reactor Core Safety Limit Lines.

B. Technical Specification Table 2.2-1 - The revised reactor core safety limit lines allow for changes in the Overtemperature ΔT and Overpower ΔT Reactor Trip System Instrumentation Setpoints. Specifically, changes to the total allowance, Z value, sensor error, K_1, K_2, K_3, K_4, K_6, nominal T_{avg} and the f_1 (ΔI) function are proposed. In addition, the phrase "by RTD Manifold Instrumentation" can be deleted because the setpoints were determined using the bounding set of operating parameters associated with either RTD bypass manifolds installed or with RTD bypass manifolds removed.

C. Technical Specification BASES 2.1.1 - Changes in the BASES reflect the use of the revised methods and correlations. Changes to the DNB design basis and new DNB design limit values are a result of the WRB-1 and WRB-2 DNB correlation and the RTDP. 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 current Vogtle LOPAR fuel is now analyzed using the WRB-1 correlation with design limit DNBR values of 1.23 and 1.22 for the typical and thimble cells, respectively.

D. Technical Specification 3/4.2.5 - The revised DNB methods and correlations and the Vogtle specific uncertainties in plant operating parameters obtained with the RTDP methodology, allow for revision of DNB-related parameters in Technical Specification 3/4.2.5. Therefore, changes are proposed for limiting values of Reactor Coolant System T_{avg}, Pressurizer Pressure, and Reactor Coolant System Flow and flow measurement uncertainty.

E. Technical Specification BASES 3/4.2, 3/4.2.2, 3/4.2.3, 3/4.2.5, and 3/4.4.1 - These BASES sections were revised to incorporate changes in the DNB design basis as a result of the new DNB correlations and RTDP methodology use.

The above proposed changes will provide additional operating and design flexibility. Specifically, the proposed Technical Specifications will accommodate higher design peaking factors ($F_{\Delta H}$), fuel rod bow, thimble plug deletion, transition core DNBR penalty, and wider axial offsets at rated thermal power associated with Relaxed Axial Offset Control.

Those transients that have DNBR as a limiting design basis criterion, and that assume reactor trips on Overtemperature ΔT and Overpower ΔT were reanalyzed with the VANTAGE-5 fuel transition analyses presented in Enclosure 4 and Appendices A and B. The safety analyses assumed the transition DNB effects from LOPAR to a full core of VANTAGE-5 fuel. The VANTAGE-5 fuel, which includes the IFM grid design feature, was generically approved by the NRC following review of the Westinghouse Topical Report WCAP-10444-P-A. The safety analyses also utilized the NRC approved RTDP methodology (Topical Report WCAP-11397-P-A), WRB-1 DNB correlation (WCAP-8762-P), WRB-2 DNB Correlation (WCAP-10444-P-A), and the improved THINC-IV model (WCAP-12330-P). Both the WRB-1 and WRB-2 DNB correlations have a DNBR limit of 1.17. However, use of the Vogtle specific RTDP calculations (WCAP-12460 and WCAP-12462, proprietary versions) resulted in the Vogtle specific DNBR design limits presented in Item C above.

Using the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and DNB correlations, were statistically combined to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values were determined such that there remains at least a 95% probability at a 95% confidence level that DNB will not occur on the most limiting fuel rod during normal operation, operational transients, and during transient conditions arising from faults of moderate frequency (Condition I and II events). The uncertainties in the plant operating parameters (pressurizer pressure, primary coolant temperature, reactor power, and reactor coolant system flow) were evaluated for Vogtle assuming two primary coolant manifold configurations. One configuration is the current Vogtle primary coolant loops with Resistance Temperatures Detector (RTD) bypass manifolds. The other configuration is a future planned plant modification with the RTD bypass manifolds eliminated and the RTD instrumentation relocated directly in primary loop thermowells. In the DNBR analyses, using the RTDP methodology, a set of plant operating parameter uncertainties was used which is bounding for operation with RTD bypass manifolds or for RTD bypass manifolds eliminated. Likewise the Technical Specifications limits in 3/4.2.5 were determined to be valid for either plant configuration. Removal of the RTD bypass manifolds is not being requested at this time. A separate submittal will be made to the NRC to allow RTD bypass manifold removal.

The safety analyses and results are discussed in more detail in Enclosure 4 and Appendix A. The results support the proposed Technical Specifications and show that the DNB design criterion is met.

Based on the information presented above and the analyses presented in the VANTAGE-5 fuel transition submittal, the following conclusions can be reached with respect to 10 CFR 50.92.

1. The proposed safety limits, reactor trip setpoints, and DNB-related parameters Technical Specifications changes do not increase the probability or consequences of an accident previously evaluated in the

FSAR. The core safety limits, trip setpoints and DNB parameters were determined using NRC reviewed and approved DNB methodologies; namely RTDP, improved THINC-IV model and the WRB-1 and WRB-2 DNB correlations. No new performance requirements are being imposed on any system or component in order to support the revised DNBR analysis assumptions. Overall plant integrity is not reduced. The DNBR sensitive transients were reanalyzed. The DNBR design criterion continues to be met. None of these changes offset parameters that could directly initiate an accident, therefore the probability of an accident has not increased. The acceptance criteria for the analyses reperformed with these revised DNB parameters continue to be met, therefore the consequences of accidents previously evaluated in the FSAR are not signifiantly changed.

2. The proposed safety limits, reactor trip setpoints, and DNB-related parameters Technical Specification changes do not create the possibility of a new or different kind of accident than any already evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes. The proposed Technical Specification changes have no adverse effects and do not challenge the performance or integrity of any safety related system. The DNBR design criterion continues to be met. Therefore, the possibility of a new or different kind of accident is not created.

3. The proposed Technical Specification changes do not involve a significant reduction in a margin of safety. The change in the DNBR design limits are associated with the use of NRC approved methodologies (RTDP, the NRC reviewed WRB-1 and WRB-2 DNB correlations and the NRC reviewed improved THINC-IV model). In addition, the VANTAGE-5 fuel design, including IFM grids, assumes use of the WRB-2 correlation and has been generically approved by the NRC. The DNB design criterion (i.e., that there is at least a 95% probability at a 95% confidence level that DNB will not occur on the most limiting rod for any Condition I or II event) remains unchanged even with the changes in DNBR design limit values. Therefore, the new DNBR design limit values associated with the DNB methodology and correlation changes, upon which the Technical Specification changes are based, do not result in a reduction in the margin of safety because the DNB design criterion continues to be met.

Based upon the preceding information, it has been determined that these proposed changes to the Technical Specifications do not involve a significant hazards consideration as defined by 10 CFR 50.92 (c).

P-11 SETPOINT

Technical Specification Table 3.3-3 specifies a P-11 setpoint of 1970 psig and an allowable value of 1980 psig. This P-11 setpoint is being changed to 2000 psig with an allowable value of 2010 psig. The proposed change provides additional operating flexibility by increasing the band between the point where safety injection is allowed to be blocked (P-11 setpoint) and the setpoint for safety injection actuation on low pressurizer pressure (1870 psig). None of the safety analyses in the VEGP FSAR use the P-11 setpoint. Therefore, there are no effects on the safety analyses as a result of this small change to the P-11 setpoint. The 30 psi increase in the difference between the P-11 setpoint and the SI setpoint will reduce the probability of an inadvertent SI signal during planned depressurization. The setpoint for the SI signal remains unaffected. During planned depressurizations, the SI signal is blocked in order to prevent an inadvertent SI actuation. The P-11 setpoint assures that the block of the SI signal is removed when pressure is returned to the normal operating range by defeating the SI block when pressurizer pressure is above the P-11 setpoint.

Based on the information presented above, the following conclusions can be reached with respect to 10 CFR 50.92.

1. The P-11 setpoint change does not increase the probability or consequences of an accident previously evaluated in the FSAR. The P-11 setpoint is not an input parameter to any transient in the FSAR. The P-11 setpoint is not an initiator for any transient. No new performance requirements are being imposed on any system or component. Consequently, overall plant integrity is not reduced. Therefore, the probability or consequences of an accident will not increase.

2. The P-11 setpoint change does not create the possibility of a new or different kind of accident from any previously evaluated in the FSAR. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the P-11 setpoint change. The P-11 setpoint change does not challenge or prevent the performance of any safety related system during plant transients. Therefore, the possibility of a new or different kind of accident is not created.

3. The P-11 setpoint change does not involve a significant reduction in the margin of safety. It is only a convenience interlock to allow SI to be blocked when below the P-11 setpoint pressure during planned cooldowns and depressurizations. The P-11 setpoint defeats the SI block when the pressurizer pressure is above the P-11 setpoint. The P-11 setpoint remains well below the initial operating pressure assumptions of the safety analyses. Therefore, the small change to the P-11 setpoint does not effect the operating envelope defined by the Technical Specifications. Therefore, the margin of safety provided by the P-11 setpoint is maintained and not reduced.

Based upon the preceding information, it has been determined that the P-11 setpoint change does not involve a significant hazards consideration as defined by 10 CFR 50.92 (c).

<u>WIDENED ACCUMULATOR WATER LEVEL RANGE</u>

The proposed change to Technical Specification 3.5.1 will widen the limiting condition for operation (LCO) defined for the range of water volume within the accumulators from a minimum of 36% span (6616 gallons) to a minimum of 29.2% span (6555 gallons) and from a maximum of 64% span (6854 gallons) to a maximum of 70.7% span (6909 gallons). The proposed Technical Specification change identified above will provide additional operating flexibility to accommodate potential changes in accumulator water level which may be experienced over an eighteen month operating cycle. Large break LOCA analyses (FSAR 15.6.5) which must account for variations in accumulator water from the nominal level have been performed as part of the VANTAGE-5 fuel transition program. The large break LOCA was reanalyzed using the NRC approved BART/BASH methodologies. The results confirm that acceptable peak clad temperatures are still achieved assuming the modified accumulator water level range with no violation of any acceptance criteria. The variation in accumulator water volume would have an insignificant effect on sump level and boron concentration.

Based on the information presented above and the analyses presented in the VANTAGE-5 fuel transition submittal, the following conclusions can be reached with respect to 10 CFR 50.92.

1. The widened accumulator water level range does not increase the probability or consequences of an accident previously evaluated in the FSAR. Accumulator water level is a parameter assumed for mitigation of the Large Break LOCA evaluated in the FSAR. Since accumulator water level is used in the role of a mitigator for this event, it does not contribute as an initiator to the probability of occurrence.

   The consequences of an accident previously evaluated in the FSAR are not increased due to the widened accumulator water level range. The radiological consequences of a Large Break LOCA have been evaluated as part of the VANTAGE-5 fuel program and are bounded by the doses currently reported in the FSAR. Therefore, the consequences to the public resulting from a LOCA previously evaluated in the FSAR have not been affected.

2. The widened accumulator water level range does not create the possibility of a new or different kind of accident than any already evaluated in the FSAR. No new accident scenarios, failure mechanisms or limiting single failures associated with the accumulator are introduced as a result of the widened accumulator water level range. The change in water level range has no adverse effect and does not challenge the performance or integrity of any other safety related system. Therefore, the possibility of a new or different kind of accident is not created.

3. The margin of safety provided by the Technical Specifications relative to the water level in the accumulators ensures that a sufficient volume of borated water will be immediately forced into the reactor core if the RCS pressure falls below the pressure of the accumulators, providing an initial cooling mechanism during large RCS pipe ruptures. The values of accumulator water level range defined by Technical Specification 3.5.1 (b) have been used in the revised LOCA analysis. The revised LOCA analysis continues to demonstrate that the

acceptance criteria are met. Therefore, the operating envelope defined by the Technical Specifications continues to be bounded by the revised analytical basis and the margin of safety provided by the revised accumulator water level range is not significantly changed.

Based upon the preceding information, it is concluded that the proposed change meets the requirements of 10 CFR 50.92 (c) and does not involve a significant hazards considerations.