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Georgia Power
the southern electric system

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NED-84-192

April 3, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC DOCKET 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
REQUEST TO CHANGE TECHNICAL SPECIFICATIONS
FOURTH CORE RELOAD

Gentlemen:

In accordance with the provisions of 10 CFR 50.90 as required by the provisions of 10 CFR 50.59(c)(1), Georgia Power Company (GPC) hereby proposes amendments to the Edwin I. Hatch Unit 2 Technical Specifications (Appendix A to the Operating License). The changes would: 1) add a new fuel type Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) curve and change the Minimum Critical Power Ratio (MCPR) curves for 8x8R and P8x8R fuel to support operation of the core design for Hatch 2 Reload 4; 2) allow operation with the new Hybrid I Control Rod Assembly; and 3) allow up to four fuel assemblies to be reloaded around the Source Range Monitor (SRM) so that the required 3 counts per second (cps) for fuel loading can be established.

The Plant Review Board has evaluated these proposed Technical Specification changes and has determined that the implementation of these proposed changes would not constitute an unreviewed safety question for the reasons stated below.

The MAPLHGR curve was generated using the methodology and acceptance criteria specified by the NRC approved licensing documents, GESSAR II and GESTAR. The MCPR curve was chosen to bound the results of the reload core design thermal limits which will be derived in accordance with the reference documents. Consequently, neither the probability of nor the consequences of an accident are increased above those analyzed in the FSAR. Additionally, the margin of safety is maintained because the acceptance criteria specified by those licensing documents is met. Because no plant system design is changed, no new type of accident or malfunction is created.

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On August 22, 1983, the NRC issued a letter reporting their review and acceptance of NEDE-22290, "Safety Evaluation of the General Electric Hybrid I control Rod Assembly." The NRC concluded, "Based on our evaluation of the information provided in (a) NEDE-22290, (b) a meeting with GE representatives, and (c) responses to NRC staff questions, we conclude that there is reasonable assurance that the substitution of Type I HICRs for other approved GE control blades will not result in unacceptable hazards to the public and should, in fact, result in improved control blade performance and a positive contribution to reactor safety. Therefore, NEDE-22290, as amended to incorporate this safety evaluation, is approved as a referential document for the GE type I HICR." Thus, this change is not an unreviewed safety issue.

The third change allows up to four fuel assemblies to be reloaded around the SRMs in order to establish the 3 cps required for SRM operability. This is very similar to the request made earlier and granted by Amendment 26 to the Unit 2 license that allowed two bundles to be loaded diagonally in order to establish SRM operability. Because the unit will have been in an outage for approximately 6 months when it is restarted, two bundles may not have been enough to establish the requisite counts on the SRM. GE spent fuel pool studies reported in Chapters 4 and 9 of GESSAR-NEDO-10741 established that any 2x2 uncontrolled array with maximum reactivity bundles will always remain subcritical (K_{∞} less than .95). The bundles that go back around the SRMs are the same bundles that left those locations. Therefore, they will remain subcritical following reinsertion because they were subcritical before they were removed. Consequently, the probability of occurrence or the consequence of an accident is not increased above those analyzed in the FSAR. Also, the margin of safety has not been reduced by using four instead of two bundles because the arrangement is significantly subcritical. Because no new mode of operation or change in plant design occurs, no new type of accident is introduced.

Included with this proposal is a determination of amendment class (attachment 7). We have determined this to be a class IV amendment. Appropriate payment is enclosed.

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Instructions for incorporation of these changes along with copies of the affected Technical Specification pages are enclosed (Attachment 3).

Pursuant to the requirements of 10 CFR 50.92, J. L. Ledbetter of the Georgia Department of Natural Resources will be sent a copy of this letter and all applicable attachments.

J. T. Beckham, Jr. states that he is 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 are true.

GEORGIA POWER COMPANY

By: J T Beckham Jr
J. T. Beckham, Jr.

Sworn to and subscribed before me this 3rd day of April, 1984.

William E. Burns
Notary Public

Notary Public, Georgia, State at Large
My Commission Expires Aug.26, 1986
DLT/mb

Enclosure

xc: H. C. Nix, Jr.
Senior Resident Inspector
J. P. O'Reilly, (NRC-Region II)
J. L. Ledbetter

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ATTACHMENT 1

NRC DOCKETS 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
REQUEST TO CHANGE TECHNICAL SPECIFICATIONS
FOURTH CORE RELOAD

Pursuant to 10 CFR 170.12, Georgia Power Company has evaluated the attached proposed amendments to Operating License NPF-5, and has determined that:

a. The proposed amendments do not require evaluation of a new Safety Analysis Report and rewrite of the facility license;

b. The proposed amendments do not contain several complex issues, do not involve ACRS review, and do not require an environmental impact statement;

c. The proposed amendments do involve more than one safety issue incorporating three changes of the Class III type;

d. Therefore, the proposed amendments result in a Class IV amendment.

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ATTACHMENT 2

NRC DOCKETS 50-366
OPERATING LICENSE NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNIT 2
REQUEST TO CHANGE TECHNICAL SPECIFICATIONS
FOURTH CORE RELOAD

Pursuant to 10 CFR 50.92, the following statements provide a summary of and the basis for the proposed changes:

1. Add a figure to define the Average Planar Linear Heat Generation Rate limit (MAPLHGR) for fuel type P8DRB284H.

   BASIS:

   Georgia Power Company proposes to add a curve of MAPLHGR vs. planar exposure for P8DRB284H fuel (Hatch 1 Reloads 5 and 6 fuel) derived on the basis of the Unit 2 system LOCA response. This change is requested in connection with the core reloading of Unit 2 to allow for introduction of a new fuel type. Fuel type P8DRB284H is a standard General Electric design as described in the NRC approved fuel licensing document, GESTAR II (NEDE-24011-P-A-6). The calculated peak clad temperatures for this fuel type correspond to the criteria specified in 10CFR50, Appendix K and in a letter from J. F. Stolz (USNRC) to J. T. Beckham (GPC), dated February 3, 1982; therefore, MAPLHGRs may be defined for exposures greater than 30,000 Mwd/st.

   The proposed MAPLHGR limits were calculated by General Electric using methods consistent with approved analyses of the loss of coolant accident and anticipated operational transients given in the Hatch-2 FSAR, and all applicable requirements stated therein are met by the proposed values. Application of the MAPLHGR limits will not result in any reduction of the margin of safety or cause any change in the consequences for postulated accidents and transients, because all acceptance criteria as defined above are met. No design changes to the plant or procedural changes are involved with this part of the amendment. Therefore, the probability of occurrence of previously considered events would remain unaffected. Because no new failure modes would be introduced, the possibility of a new type of accident would not be created.

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MAPLHGR limits for P8DRB284H are presently in the Hatch Unit 1 Technical Specifications and the fuel type has been irradicated in the Unit 1 core. As described above, no changes have been made to the acceptance criteria for the Technical Specifications or to the analytical methods used to demonstrate conformance with the Technical Specifications and regulations and the NRC has previously found the methods acceptable.

Consequently, this change is associated with a refueling and thus is consistent with Item iii of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register and will not result in a significant hazardds consideration.

2. Increase the operating limit Minimum Critical Power Ratio (OLMCPR) for 8x8R and P8x8R fuel.

BASES:

The Operating Limit MCPRs at rated core flow are defined for 8X8R and P8X8R fuel in Figures 3.2.3-1 and 3.2.3-2 of the Hatch-2 Technical Specifications. The Option B (τ =0) limits are presently 1.26 for 8X8R and 1.27 for P8X8R fuel; Option A (τ =1) limits are 1.32 for 8X8R and 1.35 for P8X8R. The parameter τ is related to the results of timing of control rod insertion speeds and is defined in Section 3.2.3 of the Technical Specifications. It is proposed to change the OLMCPRs for both of these fuel types to 1.29 for Option B and 1.37 for Option A, with linear interpolation as a function of τ.

The intent of this change is to allow for licensing the fourth Hatch-2 reload under 10CFR50.59 when final design and licensing work are completed. Based on review of the anticipated fuel mix, and on the transient analysis input parameters for the fourth reload, compared to the results of transient analyses performed for previous reloads of both Hatch units, it is judged that the proposed limits will conservatively bound the OLMCPRs that result from licensing analyses of Hatch-2 reload 4 and subsequent Hatch-2 core reloads.

Conformance with the proposed MCPR operating limits shall be assured prior to reactor startup by analyses of limiting operational transients, using the analytical methods given in the approved fuel licensing document GESTAR II (NEDE-24011-P-A-6). Application of these verified OLMCPRs will not cause any reduction of the margin of safety or produce any changes in the consequences of postualted accidents and transients because all acceptance criteria as defined above are met. No design changes to the plant or procedural changes are involved with this part of the amendment. Therefore, the probability of occurrence of previously considered events would remain unaffected. Because no new failure modes would be introduced, the possibility of a new type of accident would not be created.

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This change is requested in connection with the core reloading of Unit 2. Consequently, this change is consistent with Item (iii) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Considerations" listed on page 14870 of the April 6, 1983 issue of the Federal Register and will not result in a significant hazards consideration.

3. Change the description of the control rod assemblies in Section 5.3.2 (Design Features) of the Hatch-2 Technical Specifications to delete references to the specific materials and details of construction of the control blades.

BASIS:

This change is intended to support the use in Hatch-2 of an arbitrary number (up to 137) Type I General Electric Hybrid I Control Rod (HICR) assemblies containing some hafnium as absorber material in place of the boron carbide control rods presently in use. HICRs are intended to be standard replacement control rod assemblies for the General Electric BWR/4 D-lattice in operating reactors. The HICRs form, fit and function are identical to that of the blade it replaces. The HICR is designed to increase control rod assembly life and to eliminate cracking of absorber tubes containing boron carbide (B_4C). The essential differences between the HICR and the BWR Z-4 D-lattice control rod assemblies currently in use are:

a) improved B_4C absorber rod tube material to eliminate cracking during the lifetime of the control rod assembly, and

b) some B_4C absorber rods are replaced with solid hafnium absorber rods to increase blade life.

Other minor material and dimensional changes are described in detail in NEDE-22290-A, "Safety Evaluation of the General Electric Hybrid-I Control Rod Assembly," September 1983.

Adherence to the guidelines established for replacement of the standard B_4C control blades may require that blades in certain core locations be replaced at each refueling outage. It is expected that use of the HICR blades in these locations will allow operation of at least two 18-month fuel cycles without replacing those blades, thus reducing outage time, equipment duty and personnel exposure otherwise required for blade replacement.

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The details of design and materials for the new blades will not be included in the revised text, because those details are unnecessary and inconsistent with other portions of the Design Features Section which do not provide design or materials details. Safety design bases which must be met by control rods are enumerated in the Hatch-2 FSAR Chapter 4. Analyses documented in the approved topical report, NEDE-22290-A, have shown that those design bases are met by the HICR blades; therefore, use of these blades will cause no reduction in the margin of safety.

For example, the HICR weight and rod worth are the same as those for the currently used control rod assembly. Therefore, the scram speed and scram reactivity are also the same. It follows then that the LHGR, MCPR and MAPLHGR limits are not affected by the HICR.

Because the control rod worth is the same, the capability of the reactor to achieve the Design Basis cold shutdown reactivity margin is not affected. In addition, existing methodology for analysis of the control rod withdrawal error transient and the control rod drop accident remains valid with HICR assemblies installed. It follows then, that the probability of or consequences of all accidents and transients previously evaluated in the FSAR will not be affected by use of the HICRs.

The possibility of occurrence of an accident different than any evaluated in the FSAR is not created by use of the HICR assemblies, because there is no functional change in the control rods.

As shown above, use of the HICR assemblies in Hatch-2 does not increase the probability or consequences of a previously analyzed acccident, nor does it significantly reduce any safety margin. The result of this design change is clearly within all acceptance criteria given in the Hatch-2 FSAR as noted above.

Consequently, this change will not result in a significant hazards consideration.

4. Change the number of fuel assemblies that can be loaded around a SRM in order to assure that 3 counts per second (cps) can be achieved without the use of additional sources or dunking chambers.

<u>BASIS</u>:

The four SRM detectors are located, one per quadrant, roughly half a core radius from the center. Although these are incore detectors and thus very sensitive when the reactor us fully loaded, they lose some of their effectiveness when the reactor is partially defueled and the detectors are located some distance from the array of remaining fuel.

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GE's spent fuel pool studies, GESSAR - NEDO-10741, Chapters 4 and 9, show that: 1) sixteen or more fuel assemblies (i.e., four or more control cells) must be loaded together before criticality is possible; and 2) for an uncontrolled 2x2 array of maximum reactivity bundles, K_∞ will always be less than .95. In spiral loading sequences in the Hatch core, an array containing four or more control cells will be at most two control cells (i.e., about two feet) away from an SRM detector. The sensitivity loss in such a case is at most one decade of sensitivity (i.e., about one fifth of the SRMs logarithmic scale). This means that criticality cannot be reached during a spiral reload without an operable SRM detecting it. A sprial sequence is any sequence in which the central control cell is last unloaded and first reloaded, all fueled locations are contiguous, and no imbedded cavities or major peripheral concavities are permitted.

The Hatch 2 Technical Specifications require that the fuel assemblies be loaded into their <u>previous</u> core position next to each of the four SRMs. The loading of the bundles around the SRMs before attaining the 3 cps is permissible because these bundles were in subcritical configuration when they were removed and therefore will remain subcritical when placed back in the previous position. This request is very similar to an earlier request by Plant Hatch Unit 2 which was granted as Amendment No. 26 for loading 2 bundles next to the SRMs. Because Hatch-2 has been in an extended outage, more than 2 bundles may be required to establish the requisite 3 cps on the SRMs so that fuel loading may proceed.

The possibility of occurrence of an accident different than any evaluated in the FSAR is not created because there is no design change to any plant systems. This change does not significantly increase the probability or consequences of a previously analyzed accident because the referenced studies demonstrate inadvertent criticality with 4 bundles is not possible and further the same subcritical assemblies and arrangement that was discharged is returned to the same core location. Finally, the safety margin is not significantly reduced because the bundles remain significantly subcritical. Consequently, this change does not represent a significant hazards consideration.