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L. T. Gucwa
Manager Nuclear Safety
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SL-1263
2989N

September 22, 1986

Director of Nuclear Reactor Regulation
Attention: Mr. D. Muller, Project Director
BWR Project Directorate No. 2
Division of Boiling Water Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNITS 1, 2
TECHNICAL SPECIFICATIONS REVISIONS FOR
RWM AND RSCS OPERATION, FUEL STORAGE REQUIREMENTS,
FUEL ASSEMBLY DESIGN, MAPLHGR LIMITS, EDITORIAL CHANGES
SUPPLEMENTAL INFORMATION

Gentlemen:

The following is provided in response to your request regarding the proposed Technical Specifications change on fuel storage requirements submitted on April 15, 1986. This proposed change would remove restrictions on fuel which can be stored in the spent fuel pools at Plant Hatch to increase the flexibility of nuclear and mechanical designs for fuel assemblies. Fuel assemblies would continue to conform to all FSAR Safety Design Bases and Technical Specifications governing fuel design and spent fuel pool reactivity. The NRC staff has requested that the document in which the fuel design is analyzed to show conformance with plant design bases be specifically referenced in the Technical Specifications.

Enclosure 1 provides the page change instructions for incorporating the proposed changes and the revised Technical Specifications pages. The Bases sections of the Unit 1 and Unit 2 Technical Specifications now reference the document containing analyses on the fuel bundle which show conformance to the high density fuel rack design. For General Electric fuel, this document is the "General Electric Standard Application for Reactor Fuel" (GESTAR II).

Payment of a filing fee is not required since this letter supplies supplemental information to the April 15, 1986, submittal.

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Georgia Power Company is prepared to respond promptly to any questions you may have on this subject.

Sincerely,

L T Gucwa

L. T. Gucwa

Enclosures

GKM/tlr

c: Georgia Power Company
Mr. J. P. O'Reilly
Mr. J. T. Beckham, Jr.
Mr. H. C. Nix, Jr.
GO-NORMS

U. S. Nuclear Regulatory Commission
Dr. J. N. Grace, Regional Administrator
Mr. P. Holmes-Ray, Sr. Resident Inspector - Hatch

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ENCLOSURE 1

NRC DOCKET 50-321, 50-366
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TECHNICAL SPECIFICATIONS REVISIONS FOR
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FUEL ASSEMBLY DESIGN, MAPLHGR LIMITS, EDITORIAL CHANGES
PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS-(SUPPLEMENTAL)

The proposed revisions to the Unit 1 Technical Specifications (Appendix A to Operating License DPR-57) would be incorporated as follows:

Remove Page	Insert Page
3.10-7	3.10-7

The proposed revisions to the Unit 2 Technical Specifications (Appendix A to Operating License NPF-5) would be incorporated as follows:

Remove Page	Insert Page
B 3/4 9-2	B 3/4 9-2

3.10.A.2. Fuel Grapple Hoist Load Setting Interlocks

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 1500 lbs. in comparison to the load setting of 485 ± 30 lbs.

3. Auxiliary Hoists Load Setting Interlock

Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 485 ± 30 lb load setting of these hoists is adequate to trip the interlock when a fuel bundle is being handled.

B. Fuel Loading

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality.

C. Core Monitoring During Core Alterations

The SRM's are provided to monitor the core during periods of Unit shutdown and to guide the operator during refueling operations and Unit startup. Requiring two operable SRM's in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirements of 3 counts per second provides assurance that neutron flux is being monitored.

During spiral unloading, it is not necessary to maintain 3 cps because core alterations will involve only reactivity removal and will not result in criticality.

The loading of up to four fuel bundles around the SRM's before attaining the 3 cps is permissible because these bundles were in a subcritical configuration when they were removed and therefore they will remain subcritical when placed back in their previous positions.

D. Spent Fuel Pool Water Level

The design of the spent fuel storage pool provides a storage location for 3181 fuel assemblies in the reactor building which ensures adequate shielding, cooling, and the reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the active fuel will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the water level to less than 10 feet above the top of the active fuel. Lines extending below this level are equipped with two check valves in series to prevent inadvertent pool drainage. All fuel loaded into the Edwin I. Hatch Nuclear Plant spent fuel pool shall have an uncontrolled lattice k_∞ less than or equal to the limit for high density fuel racks described in the "General Electric Standard Application for Reactor Fuel" (GESTAR II), NEDE-24011-P-A-8. Alternatively, fuel not described in GESTAR II shall have been analyzed with another NRC approved methodology to ensure conformity to the FSAR design basis for fuel in the spent fuel racks.

E. Control Rod Drive Maintenance

During certain periods, it is desirable to perform maintenance on two control rod drives at the same time.

3/4.9.6 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.7 CRANE AND HOIST OPERABILITY

The OPERABILITY requirements of the cranes and hoists used for movement of fuel assemblies ensures that: (1) each has sufficient load capacity to lift a fuel element, and (2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.8 CRANE TRAVEL-SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel element ove irradiated fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analyses. All fuel loaded into the Edwin I. Hatch Nuclear Plant spent fuel pool shall have an uncontrolled lattice k∞ less than or equal to the limit for high density fuel racks described in the "General Electric Standard Application for Reactor Fuel" (GESTAR II), NEDE-24011-P-A-8. Alternatively, fuel not described in GESTAR II shall have been analyzed with another NRC approved methodology to ensure conformity to the FSAR design basis for fuel in the spent fuel racks.

3/4.9.9 and 3/4.9.10 WATER LEVEL-REACTOR VESSEL AND WATER LEVEL-SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 CONTROL ROD REMOVAL

This specification ensures that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.