Mr. H. L. Sumner, Jr.
Vice President - Nuclear
Hatch Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

March 23, 2000

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA5196 AND MA5197)

Dear Mr. Sumner:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 220 to Facility Operating License DPR-57 and Amendment No. 161 to Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated April 6, 1999.

The amendments revise the Technical Specifications to allow an increase of 168 fuel assemblies in the storage capacity of Unit 1's spent fuel pool and an increase of 88 fuel assemblies in the storage capacity of Unit 2's spent fuel pool.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Leonard N. Olshan, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 220 to DPR-57
2. Amendment No. 161 to NPF-5
3. Safety Evaluation

cc w/encls: See next page

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. to DPR-57
2. Amendment No. to NPF-5
3. Safety Evaluation

DISTRIBUTION:
File Center
PUBLIC
PDII-1 R/F
SCahill, RII
WBeckner,TSB
OGC
ACRS
GHill(4)

cc w/encls: See next page

DOCUMENT NAME: G:\PDII-1\HATCH\ma5196amd.wpd
To receive a copy of this document, indicate in the box C=Copy w/o attachment/enclosure E=Copy with attachment/enclosure N = No copy

No legal objection as noted

OFFICE	PDII-1/PM		PDII-1/LA	E	OGC		PDII-1/SC	
NAME	LOlshan:cn		CHawes CMH		AEH		REmch RE	
DATE	3/10/00		3/10/00		3/13/00		3/20/00	

OFFICIAL RECORD COPY

NRR-058

DF01

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 23, 2000

Mr. H. L. Sumner, Jr.
Vice President - Nuclear
Hatch Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA5196 AND MA5197)

Dear Mr. Sumner:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 220 to Facility Operating License DPR-57 and Amendment No. 161 to Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated April 6, 1999.

The amendments revise the Technical Specifications to allow an increase of 168 fuel assemblies in the storage capacity of Unit 1's spent fuel pool and an increase of 88 fuel assemblies in the storage capacity of Unit 2's spent fuel pool.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Leonard N. Olshan, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 220 to DPR-57
2. Amendment No. 161 to NPF-5
3. Safety Evaluation

cc w/encls: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 220
License No. DPR-57

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility) Facility Operating License No. DPR-57 filed by Southern Nuclear Operating Company, Inc. (Southern Nuclear), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated April 6, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-57 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 220, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: March 23, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 220

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
4.0-2	4.0-2

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.3.3 of the FSAR; and

b. A nominal 6.5 inch center to center distance between fuel assemblies placed in the storage racks.*

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 10.2.3 of the FSAR;

b. A nominal 11.5 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 203 ft 9 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3349 fuel assemblies.

* The storage rack located in the contaminated equipment storage area of the spent fuel pool shall have a nominal 6.25 inch center to center distance between fuel assemblies.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. NPF-5

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Facility Operating License No. NPF-5 filed by Southern Nuclear Operating Company, Inc. (Southern Nuclear), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated April 6, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-5 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 161, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: March 23, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 161

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains vertical lines indicating the areas of change.

Remove	Insert
4.0-2	4.0-2

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the FSAR; and

b. A nominal 6.5 inch center to center distance between fuel assemblies placed in the storage racks.*

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1 of the FSAR;

b. A nominal 11.5 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 203 ft 9 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2933 fuel assemblies.

* The storage rack located in the contaminated equipment storage area of the spent fuel pool shall have a nominal 6.25 inch center to center distance between fuel assemblies.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE DPR-57

AND AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated April 6, 1999, Southern Nuclear Operating Company, Inc. (Southern Nuclear, the licensee), et al., proposed license amendments to change the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant, Units 1 and 2. The proposed changes will allow Plant Hatch to increase the storage capacity of each unit's Spent Fuel Pool (SFP). This will be accomplished by placing a high density storage rack containing 168 storage spaces in an 8 by 21 array in the Contaminated Equipment Storage Area (CESA) of each unit's pool where currently no racks exist. The Hatch 1 SFP licensed storage capacity will increase to a total of 2933 (2845 + 88) fuel assemblies, because the licensed capacity of 2845 included the storage capacity of four original plant construction standard type storage racks (80 assemblies) that were planned for the CESA but were never installed.

2.0 EVALUATION

2.1 Occupational Radiation Exposure

The licensee plans to utilize the CESA in each unit's SFP where racks do not currently exist. The licensee estimates that the collective dose associated with the proposed fuel rack installation is in the range of 2 to 4 person-rem.

All of the operations involved in racking will utilize detailed procedures prepared with full consideration of ALARA (as low as reasonably achievable) principles.

The Radiation Protection Department will prepare Radiation Work Permits (RWPs) for the various jobs associated with the SFP rack installation operation. These RWPs will instruct the project personnel in the areas of protective clothing, general dose rates, contamination levels and dosimetry requirements. Personnel will wear protective clothing and will be required to wear personnel monitoring equipment including alarming dosimeters.

Since these license amendments do not involve the removal of any spent fuel racks, the licensee does not plan on using divers for this project. However, if it becomes necessary to utilize divers to remove any interferences which may impede the installation of the new spent fuel racks, the licensee will equip each diver with the appropriate monitoring equipment. The licensee will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and to assure that exposure is maintained ALARA.

Therefore, the staff concludes that the Hatch spent fuel storage capacity can be increased in a manner that will ensure that doses to workers will be maintained ALARA.

2.2 Solid Radioactive Waste

The necessity for pool filtration resin replacement is determined primarily by the requirement for water clarity, and the resin is normally expected to be changed about once a year. The licensee does not expect the resin change-out frequency of the SFP purification system to be permanently increased as a result of the expanded storage capacity. Overall, the licensee concludes that the additional fuel storage made available by the increased storage capacity will not result in a significant change in the generation of solid radioactive waste. The staff agrees with the licensee's conclusion.

2.3 Accident Dose Considerations

Because of the similarity between the new racks and the existing ones, and the small increase in the spent fuel capacity of the new racks, the major parameters and assumptions used in the fuel handling accident analysis are not changed and remain bounding. Therefore, the staff concludes that the increases in the capacity of the SFPs will not be accompanied by an associated increase in the radiological consequences of fuel handling accidents.

2.4 Structural Aspects

2.4.1 Storage Racks

The licensee has proposed to install two racks, a single rack for each unit, in the CESA of the SFP. The total storage capacity of two racks is 256 storage locations. The storage racks are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE). The licensee, with its contractor Holtec, performed structural analyses of the racks for the requested license amendments.

The computer program DYNARACK was used for dynamic analysis to demonstrate the structural adequacy of the spent fuel rack design under the combined effects of earthquake and other applicable loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment and are not anchored or attached to the floor or walls of the CESA. A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements and friction elements, as defined in the program, was used to simulate the three dimensional (3-D) dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

A 3-D single rack (SR) model was considered for the analyses. For the 3-D SR analyses, a rack was considered to be fully loaded and partially loaded with three different coefficients of friction (μ=0.2, 0.8 and a random value where the mean is about 0.5) between the rack pedestal and the CESA floor to investigate the fluid-structure interaction effects between the rack and the CESA walls.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration components) were generated from the design response spectra defined in the final safety analysis report (FSAR). The licensee demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra as well as matching a target power spectral density (PSD) function compatible with the design response spectra as discussed in Standard Review Plan (SRP) Section 3.7.1.

A total of sixteen 3-D SR analyses were performed. The racks were subjected to the service, upset and faulted loading conditions (Level A, B and D service limits). The results of the analyses show that the maximum displacement of the racks at the top is about 1.706 inches indicating that there is adequate safety margin against overturning of the racks. The results of the analyses also show that there is no impact potential between the rack and the CESA wall. The staff compared the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension, with corresponding allowable stresses specified in ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. The stress results show that the induced stresses under the SSE loading condition are small and all stresses in the racks are smaller than the corresponding allowable stresses specified in the ASME Boiler and Pressure Vessel Code indicating that the rack design is adequate.

The licensee also calculated the rack weld stresses at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal and cell-to-cell connections) under the dynamic loading conditions. The licensee demonstrated that all of the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code indicating that the weld connection design of the rack is adequate.

Based on: (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction and fuel loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when they are compared to the corresponding allowables provided in the ASME Boiler and Pressure Vessel Code, and (3) the licensee's overall structural integrity conclusions supported by SR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

2.4.2 Spent Fuel Storage Pool

The licensee analyzed the SFP to demonstrate the adequacy of the structures with fully-loaded fuel racks with all storage locations occupied by fuel assemblies. The fully-loaded structures were subjected to the load combinations specified in the FSAR.

The induced stresses due to two additional racks in the CESA of the SFP are smaller than the ACI 318 stress allowables. In view of the licensee's stress calculations, the staff concludes that the licensee's structural analyses demonstrate the adequacy and integrity of the structures under full fuel loading, thermal loading and SSE loading conditions. Thus, the SFP design is acceptable.

2.4.3 Fuel Handling Accident

The following two refueling accident cases were evaluated by the licensee: (1) drop of a fuel assembly with its handling tool, which impacts the baseplate (deep drop scenario) and (2) drop of a fuel assembly with its handling tool, which impacts the top of a rack (shallow drop scenario).

The results of first accident case show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads; therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The results of the second accident drop case show that damage will be restricted to a depth of 10.53 inches below the top of the rack, which is less than the acceptance criterion of 12 inches. Therefore, the staff concludes that the SFP structure and racks are adequate to withstand a fuel handling accident.

2.4.4 Conclusion

Based on the review and evaluation of the licensee's submittal, the staff concludes that the licensee's structural analysis and design of the spent fuel rack modules and the SFP structure are adequate to withstand the effects of the applicable loads including that of the SSE. The analysis and design are in compliance with the current licensing basis set forth in the FSAR and applicable provisions of the SRP, and are therefore acceptable.

2.5 Material Compatibility

2.5.1 Structural Materials

The following structural materials are used in spent fuel racks:

- All sheet metal stock consists of ASME SA240-304 stainless steel
- Internally threaded support spindle is made from ASME SA240-304 stainless steel
- Externally threaded support spindle is made from ASME SA564-630 precipitation hardened stainless steel (heat treated to 1100°F)
- Weld material is ASME Type 308

All these materials have been previously used in many other applications. They have been exposed to the environment similar or more severe than the environment in the spent fuel pools at Hatch without experiencing any observable corrosion damage. Therefore, the staff concludes that these materials are acceptable for their present application.

2.5.2 Poison Material

In the spent fuel racks, Boral is utilized as neutron absorbing material. Boral is a cermet composite material made of Type 1100 aluminum and boron carbide. The composite panel consists of boron carbide particles imbedded in Type 1100 aluminum matrix clad in Type 1100 aluminum sheets. The 1100 aluminum material imparts sufficient pitting and general corrosion resistance by forming an aluminum oxide layer on its surface when exposed to oxidizing environments. The oxide is stable in environments with a pH of 4.5 to 8.5. The boron carbide particles in Boral panels have shown good structural stability with the Type 1100 aluminum matrix material. Despite these corrosion resistant properties of Boral, some corrosion is expected. Although this will not result in a significant depletion of boron and resulting degradation of its neutron absorbing properties, some generation of hydrogen from corrosion of aluminum can occur when Boral is exposed to spent fuel pool water. This effect is more pronounced in new panels, which do not have a well formed protective oxide film. This hydrogen, when not vented, could cause swelling of the metal sheath holding, the Boral panels and resultant deformation of storage cells. In order to prevent this from occurring, the racks manufactured by Holtec will have vented Boral metal sheaths, allowing the generated hydrogen to escape. Production of this hydrogen will significantly decrease as aluminum surfaces develop a protective oxide film.

2.5.3 Conclusions

Based on its evaluation, the staff finds that the materials in the spent fuel racks, manufactured by Holtec International, are compatible with the environment in the Hatch SFPs. These racks will not undergo material degradation which could affect their ability to safely store spent and new fuel. A vented design of the metal sheaths holding the Boral panels prevents the corrosion generated hydrogen from building up pressures which could cause distortion of the fuel assembly cells. The staff concludes, therefore, that the materials used in the new spent fuel racks are acceptable.

2.6 Criticality Evaluation

The criticality analysis report that was performed in support of the additional Holtec racks was conducted in accordance with the NRC guidance contained in "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, as amended on January 18, 1979.

The boron for the Hatch 1 and 2 spent fuel racks is in the form of Boral, composed of aluminum and boron carbide. Boral as the boron containing material has been used and approved by NRC for numerous other plants. It is fastened to the fuel cells and provides a high thermal neutron removal cross section, and has proven to be structurally sound in fuel pool applications.

The current NRC-approved Hatch 1 and 2 analysis approach and the TS for the spent fuel pool and existing racks state that the reactivity status, k-effective, of the spent fuel pool shall be less than 0.95 at a 95% probability and confidence uncertainty level. The TS further indicates that this k-effective value is satisfied if the maximum k-infinity of each of the stored fuel assemblies is no greater than 1.33. This meets the NRC reactivity requirement and conforms to the requirements of General Design Criterion (GDC) 62 for prevention of criticality in fuel storage and handling and to the criterion stated in SRP 9.1.2. This criterion states that the maximum reactivity k-effective of the racks containing fuel of the highest anticipated reactivity and flooded with unborated water must not exceed 0.95. The present submittal does not propose to change this analytical approach or the spent fuel pool criterion, which remains at 0.95 for both the old and new racks. The fuel assembly chosen for the k-infinity and corresponding pool analyses was the GE fuel assemblies configuration with a 4.8 weight percent U-235 content. The GE fuel was chosen because it has the highest reactivity for a given enrichment and gadolinium loading. The 4.8 enrichment should encompass most future loadings, but this is not a requirement since the loading will have to meet the primary k-effective and k-infinity requirements.

The nuclear design and safety analysis was done by Holtec International. The primary criticality analyses of the Holtec high density spent fuel storage racks were performed with the two-dimensional multi-group transport theory computer code CASMO-4 using the 40-group cross-section library. Independent verification calculations were made with the MCNP code, a three dimensional transport theory code developed by Los Alamos National laboratory, using continuous energy cross-sections and Monte Carlo random walk technique, and the multi-group Monte Carlo code KENO5a. The KENO5a calculations used the 238-group cross-section library, which is based on ENDF/B-V data in association with the NITAWL-II program. The NITAWL-II program adjusts the uranium-238 cross-sections to compensate for resonance self-shielding effects. KENO5a is a 3-dimensional code developed by Oak Ridge National Laboratory, also using Monte Carlo, and is distributed as part of the SCALE-4.3 program package. These methodologies and cross sections are well known and have been accepted in past NRC reviews, including previous analyses by Holtec. The use of the two codes for independent verification provides greater assurance that the analysis is accurate.

The methodologies and cross-sections have been benchmarked by Holtec (and many other groups) against a number of relevant critical experiments simulating parameters related to storage racks. These benchmark calculations have been used to develop methodology bias and uncertainty factors to be added to the nominal k-effective calculations for the racks. Holtec has also determined the potential variation of rack and fuel parameters which are used in determining the k-effective of the rack and fuel system. These parameters include rack manufacturing tolerances, boron loading variations, Boral width tolerance variation, and cell lattice pitch variation. The variation of k-effective with these parameters (taken at a 95/95 probability/confidence level) was determined. These independent parameters were statistically combined with the methodology uncertainty to provide a delta k-uncertainty which was added to the base k-effective calculation. This treatment of the uncertainties is in conformance with NRC past recommendations and approvals. In addition, rack calculations were done using a conservative three-dimensional infinite arrays of cells and infinite fuel lengths.

Holtec has also investigated abnormal conditions which might be associated with the spent fuel pool. These include: (1) pool water temperature effects, (reference temperature was 20°C, but a worse case temperature, 4°C, was assumed for the investigation. The moderator temperature reactivity coefficient is negative so that any temperature increases or boiling will reduce reactivity); (2) eccentric fuel positioning (the nominal analysis case with the fuel centered in the cell yields maximum reactivity); (3) dropped fuel assembly (no significant reactivity increase); and (4) rack lateral movement (no significant reactivity increase). These analyses have provided a satisfactory demonstration that reasonably possible abnormal conditions will not lead to a reactivity problem if the required k-infinity and k-effective limits are met.

The analyses conducted by Holtec demonstrated that the criterion of k-effective being no greater than 0.95, including all the uncertainties at the 95/95 probability/confidence level, have been met. Therefore, the proposed installation of the Holtec racks at Hatch meet the NRC acceptance criteria relative to safety issues such as criticality methodology. The staff finds that the submitted analyses and the associated Technical Specifications changes to be acceptable.

The staff has reviewed the reports submitted by the licensee which describe the addition of fuel racks to the spent fuel pool, the criticality analyses performed and methods used, and the changes to the TS resulting from the analyses. The staff finds these to be acceptable and in conformance with the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

Based on this review, the staff concludes that appropriate documentation was submitted and that the proposed changes satisfy the staff positions and requirements in these areas. The criticality aspects of the spent fuel racks and the new (unburned) fuel racks are acceptable.

2.7 Thermal-Hydraulic Consideration

2.7.1 Spent Fuel Pool and Cooling System

The Unit 1 fuel pool cooling and cleanup system (FPCCS) consists of two pumps, two heat exchangers, and two filter demineralizer units. The Unit 2 FPCCS consists of one pump, one heat exchanger, and one filter demineralizer. There is an 8-inch crosstie header which allows one of the Unit 1 FPCCS trains to be shared with Unit 2 during a refueling outage. The primary safety function of the FPCCS is to transport the decay heat generated by stored spent fuel to the reactor building closed cooling system (RBCCW). FPCCS is designed to maintain the SFP bulk pool temperature to below 150 degrees Fahrenheit under normal, fuel shuffle, and full core offload conditions. These conditions are defined in Section 2.7.2 of this safety evaluation. When a full core is offloaded, one train of the residual heat removal (RHR) system can be used for cooling the SFP. The maximum design basis bulk pool temperature for the SFP is 150 degrees Fahrenheit.

Additional SFP cooling can be provided by the decay heat removal (DHR) system. The DHR system is primarily operated during refueling outages to provide decay heat removal from either spent fuel pool and allows the RHR system and/or the FPCCS to be taken out of service for inspections, repairs, or modifications. The DHR primary loop consists of two 100% capacity pumps, two plate and frame heat exchangers, and one strainer. The DHR secondary cooling loop consists of two 50% capacity pumps and two cooling towers. The DHR secondary cooling loop circulates cooling water from the basin of the cooling towers through the heat exchangers and back to the hot water side of the cooling towers. The power supplies for the DHR primary and secondary loops are from reliable power sources which can be backed up by a temporary diesel generator. The diesel generator is made available if the DHR system is to be used as the primary source of reactor core decay heat removal during the first 20 days of a refueling outage. The DHR system is sized to handle a heat load of 40 MBtu/hr. This is approximately equal to the heat load contributed to the SFP by a full-core offload 36 to 48 hours after the reactor is shutdown. According to the Hatch Unit 1 FSAR, the DHR system can maintain the SFP bulk pool temperature below 145 degrees Fahrenheit with a single failure of any DHR system component.

2.7.2 Decay Heat Load

The licensee performed decay heat load calculations in accordance with the provisions of NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling." The licensee considered three discharge scenarios which are consistent with the Plant Hatch FSAR. The discharge scenarios include normal condition, fuel shuffle, and full core offload in the refueling condition and they are defined below.

To determine the bounding case for maximum decay heat evaluation, the licensee conservatively assumed the following for each discharge scenario:

1. For all three discharge scenarios, heat loads were determined assuming the SFP was filled to maximum capacity with spent fuel discharged from 18-month operating cycles at the extended power uprate level of 2763 MWth. This assumption is conservative since Units 1 and 2 began extended power uprate operation following the Spring 1999 and Fall 1998 refueling outages, respectively. Prior to the extended power uprate level, cycles 1 through 16 for Unit 1 and cycles 1 through 12 for Unit 2 were at the initial rated power of 2436 MWth. Cycles 17 and 18 for Unit 1 and cycles 13 and 14 for Unit 2 were at the power uprate level of 2558 MWth.

2. The FPCCS heat exchangers thermal performance is based on the design maximum fouling level. Maximum design cooling water temperature of 105 degrees Fahrenheit for RBCCW was used. Both assumptions minimize the heat rejection capability of the FPCCS.

3. The heat load calculation was based on Unit 1 which has a larger spent fuel capacity than Unit 2. Therefore, the larger heat load will be used for both units.

According to the Plant Hatch FSAR, the normal condition heat load analysis is performed assuming a full spent fuel pool. The analysis showed the maximum heat load to be 8.65×10^6 Btu/hr. The heat load is calculated 52 days after offload of the last third (188 fuel bundles) of the core. At this heat load, the temperature of the SFP is maintained below 150 degrees Fahrenheit with one train of FPCCS in operation.

A fuel shuffle represents approximately one-third of the fuel bundles in the core, which are placed in the SFP. The remaining fuel bundles are shuffled in the reactor core for the next operating cycle. The licensee's heat load analysis showed the heat load to be 14.81×10^6 Btu/hr. The heat load is calculated 150 hours (6.25 days) after the reactor is shutdown. At this heat load, the temperature of the SFP is maintained below 150 degrees Fahrenheit with two trains of FPCCS or the DHR system in operation. These systems will remain in operation until the freshly discharged fuel decays to a heat load where one train of FPCCS can maintain the SFP temperature to below 150 degrees Fahrenheit. The licensee did not consider a single failure for this case. However, each system has system operating procedures in place to ensure the SFP is maintained below the design basis temperature of 150 degrees Fahrenheit.

A full core offload during a refueling outage assumed that the pool was full from fuel discharges with the last full core (560 fuel bundles) decaying for 150 hours after reactor shutdown. The calculated heat load was 34.425×10^6 Btu/hr. The licensee stated that with only the DHR system in service, SFP pool temperature will be maintained below 150 degrees Fahrenheit. As stated earlier, the DHR system is designed to maintain the SFP bulk temperature below 145 degrees Fahrenheit. Additionally, a single train of the RHR system can be aligned for fuel pool cooling and maintain the SFP temperature to below 150 degrees Fahrenheit without the assistance of either the FPCCS or the DHR system.

The staff performed confirmatory decay heat load calculations following the guidance in NRC Branch Technical Position ASB 9-2. The staff's calculations verified that the proposed decay heat loads were acceptable. The staff notes that the FPCCS is not single failure proof; however, the DHR is a reliable redundant system to the FPCCS. The staff performed an independent calculation following the guidance of the Standard Review Plan Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System." In all discharge cases, there exists a redundant method to maintain the SFP temperature below 150 degrees Fahrenheit. These redundant systems include the DHR, two trains of FPCCS, and one train of the safety related RHR system. As stated above, the DHR system can maintain the SFP bulk pool temperature to below 145 degrees Fahrenheit with a single failure of any DHR system component. The staff also verified that the long term bulk pool temperature of less than 150 degrees Fahrenheit was within the limit specified for concrete in American Concrete Institute (ACI) Standard 349. Based on our confirmatory analyses and the available systems, the staff finds the proposed decay heat loads for Plant Hatch, Units 1 and 2, acceptable.

2.7.3 Effects of SFP Boiling

In the event that all forced SFP cooling becomes unavailable, the SFP water temperature will rise and eventually reach the bulk boiling temperature of 212 degrees Fahrenheit. The licensee determined that the minimum time to reach boiling is 9.8 hours for a fuel shuffle. This assumes that the decay heat load and the bulk SFP temperature limit are at their maximum calculated values. The licensee calculated the boil-off rate at the decay heat load limit to be 15,300 lb/hr for both units. The calculated makeup rate following boiling was approximately 31 gpm for each unit. The normal source of makeup water to the SFP is the condensate water from the condensate storage tank (CST). The CST system makeup rate for Unit 1 is 390 gpm and 500 gpm for Unit 2. The safety-related source of makeup water for the SFP is the Seismic Category I plant service water (PSW). The PSW system makeup rate for both units is equal to or greater than 300 gpm. Additionally, when the reactor vessel head and the spent fuel pool gates are removed, the RHR system can be aligned to the SFP by installing two spectacle flanges and operating four isolation valves. The estimated time for this realignment is eight hours.

Based on the information provided, the staff concludes that the makeup rate to the SFP exceeds the boil-off rate. Additionally, the time in which the makeup water can be provided to the SFP is less than the minimum time-to-boil. Therefore, the staff finds that the time-to-boil analysis is acceptable.

2.7.4 Conclusions

The staff has reviewed the licensee's submittal and Plant Hatch FSAR. Based on the evaluation described above and the staff's confirmatory decay heat load calculations, the staff has concluded that the thermal-hydraulic aspects of the proposed spent fuel pool expansion are acceptable.

2.8 Control and Handling of Heavy Loads

2.8.1 Background

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides regulatory guidelines for licensees to assure safe handling of heavy loads in areas where a load drop could impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. The objectives of the guidelines are to assure that either: (1) the potential for a load drop is extremely small, or (2) the potential hazards of load drops do not exceed acceptable limits. The guidelines address criteria for establishing safe load paths; procedures for load handling operations; training of crane operators; design, testing, inspection, and maintenance of cranes and lifting devices; and analyses of the impact of heavy load drops.

In NRC's Safety Evaluation dated April 19, 1984, the staff stated that Hatch satisfied the requirements in NUREG-0612, Phase I. In addition, the licensee's response to NRC Bulletin (NRCB) 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," dated May 10, 1996, indicated that heavy loads handling operations were consistent with the guidance in NUREG-0612. It also stated that non-routine evolutions involving the movement of heavy loads (i.e., movement of spent fuel storage racks) will be evaluated in a manner consistent with the guidelines in NUREG-0612.

NRC's Safety Evaluation Report, dated March 3, 1995, approved, as part of the conversion to Improved Standard TS, the relocation of TS requirements that restrict crane travel and heavy loads movement to the licensee's Technical Requirements Manual (TRM). Section 3.9.4 of the TRM prohibits loads greater than 725 lbs. from being carried over fuel assemblies stored in the SFP. The licensee has committed to meet the TRM requirements during the rack installation and all heavy load operations. The licensee also proposes to use the defense-in-depth guidelines provided in NUREG-0612 to assure that the potential for a rack drop during the rack installation is reduced and racks are not moved over fuel in the SFP or safety-related equipment along the load path.

2.8.2 Evaluation

2.8.2.1 Crane Hoisting System/Lifting Device

The addition of the new racks in the CESA in the SFPs does not involve removal of any spent fuel pool storage racks from the SFPs. The licensee states that activities involved in installing the new additional racks will be performed in accordance with NUREG-0612 and ANSI N14.6 - 1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials." The licensee states that the heavy load limit (the actual weight of a fuel assembly and its handling tool) is 1250 lbs. It also states that the weight of a rack is 15,900 lbs. and that the maximum lifted load during the rack installation is 17400 lbs. This includes the rack, lift rig, and rigging.

The Unit 1, 125-ton Reactor Building overhead crane will be used in conjunction with a special lifting device (lifting rig) to move the new storage racks into the CESA in the SFPs of both units. The Unit 1 FSAR, Section 10.20, "Overhead Handling System," states that the reactor building crane is single-failure-proof in accordance with NUREG-0554, "Single-Failure Proof Cranes for

Nuclear Power Plants," and Section 5.1.6 in NUREG-0612. Therefore, a single failure in the Unit 1 reactor building crane will not result in the loss of the capability to safely retain its load because of (1) the high reliability of the design of the crane, and (2) the large factor of safety involved in lifting the racks. The Unit 1 FSAR also states that the cranes comply with the intent of the Crane Manufacturers Association of America (CMAA) - "Specification No. 70 for Electric Overhead Traveling Cranes," Class A1 (standby service) and the American National Standard Institute (ANSI) B30.2-1976, "Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder)," in accordance with NUREG-0612. Accordingly, load testing activities on the crane are performed prior to its initial use. Inspection and maintenance activities are performed annually and before the beginning of the rack installation.

The licensee states that a temporary hoist (lifting rig) that is remotely engaged will be attached to the overhead crane. It will be interposed between the crane hook and the rack and is specifically designed to lift the new spent fuel rack modules. It is designed and tested in accordance with the guidelines in NUREG-0612 and requirements in ANSI N14.6 -1978. It consists of four independently loaded lift rods and configured such that failure of a single rod will not result in uncontrolled lowering of the rack. Both the stress design and the load testing of the lifting rig satisfies guidelines in Section 5.1.6(1) of NUREG-0612 and Section 6 of ANSI N14.6 (1978). Accordingly, the lift rods are designed as follows: (1) with the appropriate stress design factor as specified in ANSI N14.6 (at least twice the normal stress design factors); (2) designed and load tested to 300% of the maximum weight to be lifted and suspended in air for at least 10 minutes; and (3) after load testing, the integrity and soundness of the critical weld joints are examined using a liquid penetrant. The slings are in accordance with NUREG-0612 and ANSI B30.9-1971, "Slings," and will be proof tested at a minimum of 1.5 times their rated capacity in accordance with Section 9.3.3 in ANSI B30.9.

The staff finds that the capacity of the 125-ton Reactor Building single-failure proof crane coupled with the single-failure proof lifting rig and slings will far exceed the weight of the racks and any rigging loads. Therefore, the lifting system used to move the new fuel assembly storage racks will provide a large factor of safety to enable the licensee to handle the racks with little-to-no risks of an accidental drop during rack installation.

2.8.2.2 Load Path

The licensee states that safe load paths and procedures will be developed for moving the racks into the reactor building and the CESA in the SFP. Using the Unit 1 125-ton reactor building crane, the new racks will be transferred from the reactor building airlock at elevation 158 feet up through the reactor building refueling floor equipment hatch, onto the SFP operating floor at elevation 228 feet, then into the CESA in the SFPs. Both Units 1 and 2 CESAs are separated from the rest of each pool by 8-foot concrete walls. Previously approved existing heavy load paths for casks on the operating floor will be used to access the CESAs and enable the licensee to avoid carrying the racks directly over any fuel located in the SFP. In addition, irradiated fuel assemblies will be shuffled, as needed, to allow for safe travel paths of the racks.

2.8.2.3 Heavy Loads Handling Accident Analysis

Although heavy loads' analyses are not required in accordance with Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants,' NUREG-0612," the licensee's submittal addresses the possibility of a drop of the heaviest rack module (17400 lbs.) onto the SFP liner. This satisfies the staff's recommendations in NRCB 96-02.

The licensee evaluated a 25,000 pound rack (a bounding weight) dropped vertically from a height of 40 feet above the SFP liner. The evaluation of the rack drop indicated that the SFP liner would be pierced causing leakage and local damage to the SFP concrete. However, the structural integrity of the SFP would be maintained and a rapid loss of inventory would not occur to cause any uncovering of the fuel in the storage racks. The Unit 1 FSAR, Section 10.3, "Spent-Fuel Storage," states that the liner drain system piping, which is 2 inches in diameter, is sized to limit the flow rate of any leakage through the SFP liner. Therefore, the maximum flow rate through the liner drain system is 150 gpm. In addition, the normal fuel pool water makeup system can provide 390 gpm. This is twice the maximum flow rate that any water can be drained through a breach in the liner plate. Also, makeup can be provided by the Plant Service Water System (PST). As a result, the licensee could cope with and manage damage to the SFP liner caused by a dropped rack during the rack installation.

As stated in its response to NRCB 96-02 dated May 10, 1996, because of the single failure proof capability of the Unit 1 crane, the likelihood of a load drop accident involving a dry storage cask is extremely small. Therefore, the use of a single failure proof crane precludes the need for a load drop analysis of a cask in this evolution of rack installation.

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee plans to implement measures using administrative controls and procedures to preclude load drop accidents in these four areas. The licensee will do the following: (1) provide to the rack installation crew comprehensive and plant specific training that is in accordance with ANSI B30.2, including the use of procedures on the use of the lifting system, upending equipment, and all other aspects of the rack installation; (2) use redundantly designed lifting rigs; (3) perform inspection and maintenance checks on the cranes and lifting devices prior to the rack operation; and (4) use specific procedures that cover the entire rack installation effort, including the identification of required equipment, inspection, acceptance criteria prior to load movement, defining safe load paths, and steps and precautions for proper load handling and movement.

The staff accepts the licensee's finding that SFP integrity would not be breached if a rack drop was to occur. Also, the staff agrees with the licensee that the use of the crane in conjunction with administrative procedures and controls focused on, but not limited to, the areas noted above will enable the licensee to maintain safety during the rack installation.

2.8.3 Conclusion

Based on the preceding discussions, the staff finds that the aforementioned considerations of heavy loads to support the proposed changes to TS 4.3 and the increase in the storage capacity for spent fuel assemblies in the SFPs' CESA are acceptable. The licensee's use of the Unit 1 reactor building 125-ton single failure proof crane, in conjunction with administrative controls that are in accordance with NUREG-0612, will help to maintain safety during the installation of the additional new spent fuel storage racks in the SFPs' CESA. These considerations for moving heavy loads which will enable the licensee to move the racks during installation while preventing any damage to spent fuel and the SFP structure, are acceptable to the staff.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION)

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on March 23, 2000 (65 FR 15661) for this amendment. Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: R. Tadesse
A. Attard
Y. Kim
K Parczewski
B. Thomas
K. Kavanaugh
J. Lee

Date: March 23, 2000