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Docket Nos. 50-321
and 50-366

Mr. Charles F. Whitmer
Vice President - Engineering
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Whitmer:

The Commission has issued the enclosed Amendment No. 74 to Facility Operating License No. DPR-57 and Amendment No. 15 to Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, respectively. These amendments consist of changes to the Technical Specifications in response to your application dated July 9, 1979, as amended July 27, 1979, September 21, 1979, October 29, 1979, November 30, 1979, December 31, 1979, and February 18, 1980.

These amendments authorize the installation and use of new high density storage racks for the storage of spent fuel assemblies in the spent fuel storage pool.

Copies of the Safety Evaluation, Environmental Impact Appraisal, and the Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 74 to DPR-57
2. Amendment No. 15 to NPF-5
3. Safety Evaluation
4. Environmental Impact Appraisal
5. Notice

cc w/enclosures:
See page 2

Concur as to form only. It is noted that the dates of supplements in the notice are not the same as those in the safety eval. 4/10

corrected 4/15/80

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DATE	4/9/80	4/9/80	4/9/80	4/11/80	4/16/80	

NRC FORM 318 (9-76) NRCM 0240 ☆U.S. GOVERNMENT PRINTING OFFICE: 1979-289-369 cc:

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

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 NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 74
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated July 9, 1979, as amended July 27, September 21, October 29, November 30, December 31, 1979 and February 18, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations 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;

   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 amended by 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 Appendices A and B, as revised through Amendment No. 74, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

W. P. Gammill

W. P. Gammill, Acting Assistant Director
for Operating Reactor Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 21, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 74

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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

Remove Page	Insert Page
3.10-5	3.10-5
3.10-7	3.10-7
3.10-9	3.10-9
3.10-10	3.10-10
5.0-1	5.0-1
5.0-2	5.0-2

4.10.F.3 Monorail Hoist (Continued)

a. The trolley and hoist shall be demonstrated to be operable by a trial lift of the spent fuel pool gate or an equivalent weight.

b. A visual inspection shall be made to insure the structural integrity of the 5-ton monorail hoist.

3.10.G <u>Spent Fuel Cask Lifting Trunnions and Yoke</u>

See note for Specification 3.10.F.1 above.

3.10.H <u>Time Limitation</u>

Irradiated fuel shall not be handled in or above the reactor prior to 24 hours after reactor shutdown.

3.10.I <u>Crane Travel-Spent Fuel Storage Pool</u>

Loads in excess of 1600 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage racks.

4.10.G <u>Spent Fuel Cask Lifting Trunnions and Yoke</u>

See note for Specification 4.10.F.1 above.

4.10.I <u>Crane Travel-Spent Fuel Storage Pool</u>

Loads, other than fuel assemblies or control rods, shall be verified to be ≤ 1600 pounds prior to movement over fuel assemblies in the fuel storage pool racks.

3.10.A.2. Fuel Grapple Ho' ' Lo._ 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 sprial 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 diagonally adjacent 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 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 10CFR20. 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.

E. Control Rod Drive Maintenance

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

3.10.H Time Limitation

The radiological consequences of a fuel handling accident are based upon the accident occurring at least 24 hours after reactor shutdown.

3.10.I Crane Travel-Spent Fuel Storage Pool

A maximum weight of 1600 pounds shall be permitted to be transported over stored spent fuel in order to minimize the consequences of a load handling accident.

3.10.J References

1. FSAR Section 7.6, Refueling Interlocks

2. FSAR Section 7.5, Neutron Monitoring System

3. Morgan, W. R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors," General Electric Company, Atomic Power Equipment Department, November 1968, revised April 1969 (APED-5706)

4. FSAR Section 10.3, Spent Fuel Storage

5. FSAR Section 3.6.5.2, Reactivity Control

4.10. REFUELING

A. Refueling Interlocks

Complete functional testing of all refueling interlocks before any refueling outage will provide positive indication that the interlocks operate in the situations for which they were designed. By loading each hoist with a weight equal to the fuel assembly, positioning the refueling platform, and withdrawing control rods, the interlocks can be subjected to valid operational tests. Where redundancy is provided in the logic circuitry, tests can be performed to assure that each redundant logic element can independently perform its functions.

C. Core Monitoring During Core Alterations

Requiring the SRM's to be functionally tested prior to any core alteration assures that the SRM's will be operable at the start of that alteration. The daily response check of the SRM's ensures their continued operability.

D. Spent Fuel Pool Water Level

A daily record of the Spent Fuel Pool Water Level to determine that the minimum of 8.5 feet is not exceeded is considered sufficient to ensure that radiological shielding is maintained.

E. Control Rod Drive Maintenance

Refueling interlocks and core monitoring surveillance are discussed in 4.10.A and 4.10.C above. The choice of the strongest (highest reactivity worth) rod which will be used for a determination of the relevant shutdown margins is based on prior core calculations supplemented by empirical data obtained from similar cores. From similar data and calculations the reactivity worth of rods adjacent to a withdrawn rod will also be known. Thus the surveillance shutdown margins can be evaluated in terms of rod position.

F. Reactor Building Cranes

Modifications to the main reactor building crane are being studied in order to increase its ability to withstand a single failure. A spent fuel cask will not be lifted until these modifications have been accepted by the AEC and the AEC has approved the lifting of casks by the crane and the appropriate Technical Specifications.

G. Spent Fuel Cask Lifting Trunnions and Yoke

See note for Bases 4.10.F above.

I. Crane Travel-Spent Fuel Storage Pool

Refer to Bases 3.10.I.

5.0 MAJOR DESIGN FEATURES

A. Site

Edwin I. Hatch Nuclear Plant Unit No. 1 is located on a site of about 2244 acres, which is owned by Georgia Power Company, on the south side of the Altamaha River in Appling County near Baxley, Georgia. The Universal Transverse Mercator Coordinates of the center of the reactor building are: Zone 17R LF 372,935.2m E and 3,533,765.2m N.

B. Reactor Core

1. Fuel Assemblies

The core shall consist of not more than 560 fuel assemblies of the licensed combination of 7x7 bundles which contain 49 fuel rods and 8x8 fuel bundles which contain 62 or 63 fuel rods each.

2. Control Rods

The reactor shall contain 137 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% of its theoretical density.

C. Reactor Vessel

The reactor vessel is described in Table 4.2-2 of the FSAR. The applicable design specifications shall be as listed in Table 4.2-1 of the FSAR.

D. Containment

1. Primary Containment

The principal design parameters and characteristics of the primary containment shall be as given in Table 5.2-1 of the FSAR.

2. Secondary Containment

The secondary containment shall be as described in Section 5.3.3.1 of the FSAR and the applicable codes shall be as given in Section 12.4.4 of the FSAR.

3. Primary Containment Penetrations

Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.3.4 of the FSAR.

E. Fuel Storage

1. Spent Fuel

All arrangements of fuel in the spent fuel storage racks shall be maintained in a subcritical configuration having a k_{eff} not greater than 0.95.

2. New Fuel

The new fuel storage vault shall be such that the k_{eff} dry shall not be greater than 0.90 and the k_{eff} flooded shall not be greater than 0.95.

3. Fuel Storage

Fuel in the Spent Fuel Pool shall have a maximum fuel loading of 15.2 grams of Uranium-235 per axial centimeter.

5.0.F. Seismic Design

The reactor building and all engineered safeguard systems are designed for the design basis earthquake with a horizontal ground acceleration of 0.15 g. The operating basis earthquake has a horizontal ground acceleration of 0.08 g.

G. References

1. FSAR Section 4.2, Reactor Vessel and Appurtenances Mechanical Design
2. FSAR Section 5.2, Primary Containment System
3. FSAR Section 5.3, Secondary Containment System
4. FSAR Section 12.4.4, Governing Codes and Regulations
5. FSAR Section 10.3, Spent Fuel Storage
6. FSAR Section 10.2, New Fuel Storage

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

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
License No. NPF-5

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated July 9, 1979, as amended July 27, 1979, September 21, October 29, November 30, December 31, 1979 and February 18, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations 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;

   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 amended by 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 Appendices A and B, as revised through Amendment No. 15, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

W. P. Gammill

W. P. Gammill, Acting Assistant Director
for Operating Reactor Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 21, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 15

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

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

Remove Page	Insert Page
5-4	5-4

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 185 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2845 fuel assemblies.

FUEL STORAGE

5.6.4 Fuel in the Spent Fuel Pool shall have a maximum fuel loading of 15.2 grams of Uranium-235 per axial centimeter.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

8005130090

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 74 TO LICENSE NO. DPR-57

AND AMENDMENT NO. 15 TO LICENSE NO. NPF-5

GEORGIA POWER COMPANY, ET AL.

EDWIN I. HATCH NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NO. 50-321 AND 50-366

1.0 Introduction

By application dated July 9, 1979, and supplements thereto dated July 27, September 21, October 29, November 30, December 31, 1979 and February 18, 1980, the Georgia Power Company (the licensee) proposed to change the spent fuel pool (SFP) storage design for E. I. Hatch Nuclear Plant Units 1 and 2 (Hatch 1/2) from the design which was reviewed and approved in the operating license review and described in the FSAR. The proposed change consists of increasing the total spent fuel storage capacity of the SFP from 840 fuel assemblies to 3171 fuel assemblies for Hatch 1 and from 1120 fuel assemblies to 2755 fuel assemblies for Hatch 2. The increase will be accomplished within the existing spent fuel pool by the installation of spent fuel storage racks which utilize fixed neutron absorbers to allow higher density storage.

2.0 Discussion

The proposed spent fuel assembly racks are to be made up of alternating stainless steel containers. Thus, there will be only one container wall between adjacent spent fuel assemblies. Each container wall is to have a sheet of Boral sandwiched between stainless steel, 0.0355 inch inside and 0.090 inch outside. The containers will be about 14 feet long and will have a square cross section with an outer dimension of 6.653 inches and a total wall thickness of 0.2015 inches. The nominal pitch between fuel assemblies will be 6.563 inches, which when combined with the fuel region dimension of 5.12 inches gives an overall fuel region volume fraction of 0.61.

The approximately 6.25 inch wide Boral sheets, which are to be in every container wall, are made up of a central segment of a 0.056 inch thick dispersion of boron carbide in aluminum, which is clad on both sides with 0.010 inches of aluminum. GPC stated in its July 9, 1979 submittal that the minimum, equivalent homogeneous, areal concentration of boron will be 0.013 grams of the boron ten isotope per square centimeter of Boral plate.

2.1 Criticality Analyses

The General Electric Company (GE) performed the criticality analyses for GPC. GE made the calculations with the MERIT Monte Carlo program with cross sections which were processed from ENDF/B-IV data. The accuracy of this calculational method was assessed by using it to calculate representative critical experiments. From this qualification program, GE determined that this calculational method underpredicts K_{eff} by 0.5 percent Δ.

GE used these computer programs to calculate the neutron multiplicatior factor for an infinite array of fuel assemblies in the nominal storag lattice at 20°C. The calculational model of the fuel assembly which GE used for these spent fuel pool calculations had discrete, unirradiated fuel pins with varying amounts of uranium-235 in them with a total fuel loading of 15.2 grams of uranium-235 per axial centimeter of fuel assembly. There was no burnable poision included in any of the fuel pins. The Boral plates, which were also discretely represented in the calculational model, had the quoted minimum concentration of boron in them, i.e., 0.013 grams of boron - ten per square centimeter of plate. For the minimum possible as-built pitch (i.e., 6.503 inches) GE's calculated value for this storage lattice k_∞ is 0.87. GE then calculated the k_∞ 's for the following conditions: (1) increasing the temperature to 65°C; (2) increasing the lattice pitch; (3) locating every four fuel assemblies as close together as possible; and (4) reducing the density of the water. GE found that all of these changes resulted in a decrease in k_∞ .

Because of the alternating lattice design, wherein there will be only one storage container for every two fuel assemblies, there will be spaces on the periphery of the rack modules which will not have Boral plates. Thus, it will be possible for two rack modules to be put together so that adjacent fuel assemblies will not have a Boral plate between them. GE calculated the effect of these missing Boral plates for the minimum attainable gap between rack modules and found that it would not increase the maximum k_∞ of 0.87. GE also analyzed the situation where the defective fuel storage spaces, which are attached externally to some of the storage modules, are all filled with fuel assemblies. GE states that these analyses have demonstrated that the k_{eff} is <0.95.

In regard to an onsite neutron attenuation test to verify the presence of the boron plates, GPC states the following:

> "The presence of the neutron absorber material in the fabricated fuel storage module will be verified at the reactor storage pool site by scanning each storage tube in the modules with a neutron source and neutron detectors. The recorded results provide a comparison between neutron absorption through each Boral sheet and neutron absorption measured through the stainless steel without Boral. A significant increase in neutron absorption verifies the presence of Boral."

2.1.1 EVALUATION

GE's use of a non-uniform distribution of fuel in the criticality calculations for the spent fuel pool results in a lower value of the neutron multiplication factor than would have been obtained with a uniform fuel distribution. If a uniform distribution of fuel had been used, the calculated value of the neutron multiplication factor would have been about 0.91 instead of 0.87. The maximum possible neutron multiplication factor in these pools is higher than 0.87. This maximum value is set by the composition of fuel assemblies which are placed in the defective fuel storage locations and in the adjacent locations.

By assuming new, unirradiated fuel with no burnable poison or control rods, these calculations yield the maximum neutron multiplication factor that could be obtained throughout the life of the fuel assemblies. This includes the effect of the plutonium which is generated during the fuel cycle.

Because the neutron multiplication factor is not measured in spent fuel pools, the only available value is a calculated one. This calculation is complex, and it has many inputs and possible uncertainties. Thus, the NRC staff is required to review these calculations to determine that the uncertainties in the calculated neutron multiplication factor are not excessive. Accordingly, we have reviewed the calculations which were made for the fuel assemblies which are described in this submittal. We find that all factors that could affect the neutron multiplication factor of the fuel assemblies, which are described in this submittal, in the spent fuel pools with the proposed racks have been accounted for and that the neutron multiplication factor will not exceed 0.95. This is NRC's acceptance criterion for the maximum (worst case) calculated neutron multiplication factor in a spent fuel pool. Accordingly, there is a technical specification which limits the neutron multiplication factor (k_{eff}) in the spent fuel pool to 0.95.

GPC may want to store fuel assemblies other than those described in this submittal in the proposed storage racks. Since the maximum neutron multiplication in the pool could be determined by the composition of these other fuel assemblies, the NRC required an additional technical specification to preclude any unreviewed increase, or increased uncertainty, in the calculated value of the neutron multiplication factor which could raise the actual neutron multiplication factor in the fuel pool above 0.95 without being detected. An acceptable form of this additional technical specification is to limit the fuel loading of any assembly stored in the proposed racks to that of the fuel assembly which was modeled for the calculations in this submittal.

In regard to G C's onsite neutron attenuation testing of the Boral plates, we f .d that with the quality assurance program procedures in effect there should be no Boral plates missing from the prescribed locations in the fabricated fuel storage modules. If GPC finds any Boral plates missing it shall specifically note and document this finding in its test report, and report it to the NRC.

2.1.2 CONCLUSION

We find that when any number of the fuel assemblies, which GPC described in these submittals, which have no more than 15.2 grams of uranium-235, or equivalent, per axial centimeter of fuel assembly, are loaded into the proposed racks, the k_{eff} in the fuel pool will be less than the 0.95 limit. We also find that in order to preclude the possibility of the k_{eff} in the fuel pool from exceeding this 0.95 limit without being detected, it is necessary pending an NRC review, to prohibit the use of these high density storage racks for fuel assemblies that contain more than 15.2 grams of uranium-235 per axial centimeter of fuel assembly. On the basis of the information submitted, and the k_{eff} and fuel loading limits stated above, we conclude that the health and safety of the public will not be endangered by the use of the proposed racks.

2.2 SPENT FUEL COOLING

The licensed thermal power for each of the two Hatch reactors is 2436 MWth. GPC plans to refuel these reactors annually at which times about 140 of the 560 fuel assemblies in each core will be offloaded. To calculate the maximum heat loads in the spent fuel pool GPC assumed a 150 hour (6.25 day) time interval between reactor shutdown and the time when either the 140 fuel assemblies in the normal refueling or the 560 fuel assemblies in the full core offload are placed in the spent fuel pools. For this cooling time GPC used the method given in the NRC Standard Review Plan 9.2.5 to calculate maximum heat loads of 11.6×10^6 BTU/hr for twenty two successive annual refuelings and 28.7×10^6 BTU/hr for the full core offload which fills the pool after nineteen annual refuelings. This calculation was made for Unit 1 which is to have the larger amount of spent fuel stored in it and consequently the greater heat load.

There are three trains of spent fuel cooling at the E.I. Hatch Nuclear Plant. There is one train for each of the two units and another one, called a swing train, which can cool either of the two spent fuel pools. Each cooling train consists of a pump and a heat exchanger. The FSAR for Unit 1 states that the pumps for its cooling train and the swing train are designed to pump 610 gpm (3.05×10^5 pounds per hour) and that each of the two heat exchangers is designed to transfer 4.25×10^6 BTU/hr from 125°F fuel pool water to 105°F Reactor Building Component Cooling Water (RBCCW), which is flowing through the shell side of the heat exchanger at the rate of 1200 gpm (6.0×10^5 pounds per hour). The FSAR for Unit 2 states that its spent fuel pool cooling pump is designed to pump 650 gpm (3.25×10^5 pounds per hour) and that its heat exchanger is designed to transfer 4.25×10^6 BTU/hr from 125°F fuel pool water to 105°F RBCC water which is flowing through the shell side of the heat exchanger at the rate of 1200 gpm.

GPC states that this system, with two trains operating, for either of the two spent fuel pools, will be able to keep the outlet water temperature at or below 133°F through the final annual refueling. For cooling an offloaded full core GPC states that a single train of the Residual Heat Removal (RHR) system, when aligned to either of the fuel pools, will, by itself, maintain the spent fuel pool outlet water temperature at or below 145°F. GPC states that, when the reactor vessel head and the spent pool gates are removed, the RHR system can be aligned to the spent fuel pool by installing two spectacle flanges and operating four isolation valves. The estimated time for this realignment is eight hours.

In regard to emergency make up water for the spent fuel pools, GPC states that the water level in the fuel pools will be maintained by the Seismic Category I Plant Service Water system in each unit.

2.2.1 EVALUATION

Using the method given on pages 9.2.5-8 through 14 of the NRC Standard Review Plan, with the uncertainty factor, K, equal to 0.1 for decay times longer than 10^3 seconds, we find that GPC's calculated values for the maximum peak heat loads in the modified spent fuel pools are conservatively high. From our calculations we also find that the maximum incremental heat load that could be added by increasing the number of spent fuel assemblies in the Unit 1 pool from 840 to 3181 will be 2.3×10^6 BTU/hr. This is the difference in peak heat loads for full core offloads that essentially fill the present and the modified pools. Since more spent fuel is to be stored in the Unit 1 pool, the incremental heat load for the Unit 2 pool will be smaller.

We calculate that with two trains in operation, the spent fuel pool cooling system of Unit 1 can maintain the fuel pool outlet water temperature below 133°F for a peak annual refueling heat load of 11.6×10^6 BTU/hr. Since the Unit 2 pool will have a smaller heat load its outlet water temperature will be less. We find that when the RHR system of Unit 1 is aligned with its spent fuel pool it will, by itself, have sufficient capacity to keep the temperature of the outlet water below 150°F for a peak, full core, heat load of 28.7×10^6 BTU/hr.

Since the peak heat load for Unit 2 is somewhat smaller, its RHR system can also kee' the outlet water below 150°F.

Because, as stated in GPC's submittal, the Residual Heat Removal System (RHR) and the spent fuel pool cooling system piping that is exposed to RHR flow are designed to Seismic Category I criteria, the cooling of a full core offload in the spent fuel pool would not be precluded due to a seismic event. Thus, the worst credible accident for spent fuel cooling at either Hatch Units 1 or 2 is the complete loss of cooling after a normal refueling and a resumption of power operation. For this situation the heat up rate of the water in either of the spent fuel pools will be less than 5°F/hr. Assuming a maximum fuel pool temperature of 133°F there will be a minimum time period of 16 hours before bulk pool boiling commences. After this the maximum boil off rate will be 24 gpm.

We find that sixteen hours will be sufficient for GPC to establish a 24 gpm make up rate from the Unit's Seismic Category I Plant Service Water system. We also find that under bulk boiling conditions the temperature of the fuel will not exceed 350°F. This is an acceptable temperature from the standpoint of fuel element integrity and surface corrosion.

2.2.2 Spent Fuel Pool Cleanup System

The SFP cooling and cleanup system consists of a filter vessel, a resin trap, a holding pump, a precoat mixing tank and pump and the required piping, valves and instrumentation. The entire fuel pool cooling water flow is processed through the filter/demineralizer system. The normal flowrate will produce approximately four complete water changes per day of the fuel pool. The filter is enclosed in a shielded cell to keep exposures of plant personnel to minimum levels of radiation.

Because we expect only a small increase in radioactivity released to the pool water as a result of the proposed modification as discussed in Section 4.1, we conclude that the SFP purification system will keep concentrations of radioactivity in the pool to levels which have existed prior to the modification.

2.2.3 Conclusion

We find that the present cooling capacities in the spent fuel pools of the Edwin I Hatch Nuclear Plant will be sufficient to handle the incremental heat loads that will be added by the proposed modifications. We also find that these incremental heat loads will not alter the safety considerations of spent fuel pool cooling from that which we previously reviewed and found to be acceptable. We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the use of the proposed design.

2.3 Installation of Racks and Fuel Handling

There are presently no fuel assemblies in the E.I. Hatch Unit 2 pool. It is GPC's intention to install the new racks without having fuel assemblies in the Unit 2 pool. There are about 260 spent fuel assemblies in the Unit 1 pool, but the two pools at Hatch are connected by a fuel transfer canal. Thus, after the new racks are installed in the Unit 2 pool the spent fuel assemblies in the Unit 1 pool can be transferred to the new racks in the Unit 2 pool prior to installing the new racks in the Unit 1 pool. Thus, both of these rack changes can be done without fuel assemblies in the pool.

2.3.1 Evaluation

Since there will be no fuel assemblies in the fuel pool during the modification, it will not be possible for an accident to result in any increased neutron multiplication factor. After the racks are installed in the pools, the fuel handling procedures that will be implemented in and around the pool will be the same as those procedures that were in effect prior to the modifications. These were previously reviewed and found acceptable by the NRC.

2.3.2 Spent Fuel Handling

The NRC staff has under way a generic review of load handling operations in the vicinity of spent fuel pools to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. Because the main crane meets the requirements of NUREG-0554 we have concluded that the likelihood of a heavy load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of the SFP are necessary while our review is under way.

The consequences of fuel handling accidents (i.e., rupture of all the fuel pins in the equivalent of one fuel assembly and the subsequent release of the radioactive inventory within the gap) in the spent fuel pool are not changed from those presented in the Safety Evaluation (SE) dated June 13, 1978.

2.3.2 Conclusion

We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the installation and use of the proposed racks.

2.4 Structural and Mechanical Design

Georgia Power Company proposes to replace existing spent fuel storage racks at its Edwin I. Hatch Nuclear Plant Units 1 and 2 with racks of increased capacity. The present Hatch 1 and 2 spent fuel pools contain racks that can hold 840 and 1120 fuel assemblies, respectively. The replacement of high density spent fuel storage racks will provide 3171 storage spaces in Hatch 1 and 2755 in Hatch 2. The proposed modification will provide storage capacity up to the year 1997 with a full core reserve, assuming annual quarter reloads. The replacement spent fuel storage racks are to be fabricated primarily from type 304 stainless steel. The individual fuel assemblies will be stored in square fuel storage cells formed from an inner shroud of stainless steel, a center sheet of boral and an outer shroud of stainless steel. The cells act as a storage space and provide neutron absorption from the boral sheet to allow spacing of fuel in a 6.5 inch by 6.5 inch array. The outer and inner tubes are welded together after being sized to the required dimensional tolerances. The completed storage tubes are fastened together by angles welded along the corners and attached to a base plate to form storage modules.

The base plate of each module is supported on all four corners by 2-inch thick foot pads. The foot pads rest on 6-inch thick corner-support pads which in turn rest on the fuel pool floor liner. This raises the base plate of the module a minimum of 8 inches above the floor of the fuel pool, allowing sufficient clear area to permit natural circulation of cooling water to the modules without taking credit for sources of forced cooling.

2.4.1 Evaluation

The proposed modification for the spent fuel storage capacity expansion program has been reviewed in accordance with the NRC report "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 1978. The structural and mechanical review consisted of an examination of the following areas: the proposed design criteria, the design loads and load combinations, methods of analysis, the dropped fuel assembly accident, the material properties and allowable stresses of type 304 stainless steel, the hydro-dynamic effects, the fabrication and installation provisions, and the effect of increased loads on the floor slab and liner.

The high density spent fuel module has been analyzed for both OBE and SSE conditions. A damping ratio of 2 percent of the critical was used in the analysis for the SEE condition and 1 percent for the OBE condition without any added damping from fluid effects. Seismic spectra are based on Hatch 2 which bound the spectra for Hatch 1. Combination of the modal response and the effect of the three components of an earthquake has been performed in accordance with the applicable provisions of USNRC Regulatory Guide 1.92. The total mass of the module including the hydrodynamic effect, which represents the inertial properties of the fluid surrounding the submerged modules, was calculated and then used to perform the seismic analysis. Stress analyses were done for both OBE and SSE conditions based upon the shears and moments developed in the finite-element dynamic analysis of the seismic response. These values were then compared and found to be within the allowable stresses referenced in ASME Section III, Subsection NF.

Seismic loads obtained from the response spectra analysis were increased by impact factors which account for the effects of the clearance gap between the storage cells and the fuel assemblies contained therein.

The sliding analysis was done using the two-dimensional, non-linear DRAIN-2D and SEISM computer codes. DRAIN-2D was originally developed at the University of California at Berkely, SEISM was developed by GE. Both computer codes meet NRC-QA requirements. Sliding and overturning of the module were studied for the SSE and OBE conditions. All of the modules were found to be stable under the worst postulated seismic loading conditions, and the minimum 2-inch clearance between modules precluded contact during a seismic event.

The spent fuel pool structure was re-evaluated based on the increased loads caused by the new high density spent fuel storage racks. The ACI code 349-76 "Code Requirements of Nuclear Safety Related Concrete Structure" as supplemented by USNRC Regulatory Guides and positions was used as the design basis for the structural re-evaluation.

A three-dimensional methematical model was developed for each spent fuel pool structure. Each mathematical model is composed of plate shell elements, beam elements, truss elements, and boundary elements to idealize the existing structure. Structural properties for the elements were selected based on in-situ conditions.

The re-evaluation for the Unit 1 spent fuel pool structure showed that the existing structure and liner plate would have adequate capacity to carry the additional loads imposed by the high density spent fuel racks.

Fuel assembly drop accidents were analyzed using analytical methods in accordance with the "Operating Technical Position for Review and Acceptance of Spent Fuel Storage and Handling Applications". In estimating local damages in the module, the maximum strain energy resulting from plastic deformation was equated to the maximum potential energy of the fuel. Energy dissipation attributable to the viscosity of the water and plastic deformation of the fuel bundle was ignored for conservative results. The stainless steel for the module was assumed to exhibit a bi-linear hystresis relationship, with yield stress and ultimate stresses as the two control points.

A free fall of a fuel assembly into the fuel pool liner was evaluated to serve as a basis for concluding that the leak tightness of the fuel pool liner plate is maintained. The evaluation demonstrated that the energy developed by a freely falling fuel assembly from a height extending 27 inches above a module would not cause liner plate perforation.

Specifications which impose quality control requirements during the design, procurement, fabrication, installation, and testing of the storage system were developed specifically for the high density spent fuel. Periodic audits of the various facilities and practices are performed by certified quality assurance personnel to ensure that these Quality Assurance/Quality Control requirements are being met. All welding and nondestructive examinations will be done in accordance with the applicable provisions of the ASME Boiler & Pressure Vessel Code, Section IX, and the American Society for Nondestructive Testing.

2.4.2 Material Considerations

The Type 304 stainless steel used in the new spent fuel storage racks is compatible with the storage pool environment, which is demineralized water. Based on our review of previous operating experience with similar materials approved in use, we have concluded that there is reasonable assurance that no significant corrosion of the racks, the fuel cladding, or the pool liner will occur over the lifetime of the plant. The aluminum in the Boral neutron absorber plates will experience some galvanic corrosion with the stainless steel tubes encapsulating the Boral being vented to the pool water environment, although in the high resistivity pure water environment any galvanic action will be minimized. The more noble stainless steel will not be affected by any galvanic attack when contacted with aluminum. Although slight galvanic corrosion may occur in the aluminum of the Boral plates, it should not have any significance on the neutron absorption capability of the Boral and certainly no effect on storage rack structural integrity for a period in excess of 40 years.

2.4.3 Conclusion

The structural, mechanical and material aspects of the spent fuel storage racks have been evaluated based upon NRC guidance provided in the report entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 1978. Based upon our review of the analyses and the design done by the licensee, we conclude that the rack structure itself, the supporting pool liner and slab, are capable of supporting the applied loads without exceeding relevant stresses of Subsection NF or the FSAR Design Criteria. As previously stated, we find the material fabrication, installation, and examination criteria acceptable. We conclude that the proposed modifications to the Edwin Hatch spent fuel storage are in conformance with NRC requirements.

2.5 Occupational Radiation Exposure

We have reviewed the licensee's plan for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for this operation is estimated by the licensee to be about 14 to 28 man-rem as explained below.

We consider this to be a reasonable estimate because it is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification. The licensee considered the number of individuals performing a specific job, their occupancy time while performing this job, and the average dose rate in the area where the job was being performed.

The modi ication will first be performed in Hatch 2, which is at the present time dry, but would proceed in the same manner if wet. After completion of the reracking of Hatch 2, the spent fuel presently stored in Hatch 1, will preferentially be transferred to Hatch 2 through the transfer canal, or may be concentrated away from rerack work locations. In either case, the work to be performed for Hatch 1, will be performed in a manner consistent with as low as is reasonably achievable (ALARA) occupational exposure. The licensee will remove unnecessary radioactive equipment and material from the Hatch 1 fuel pool prior to the installation work, and will pre-plan procedures necessary for the removal of the old racks and installation of the new ones. The licensee does not anticipate the need for divers at this time. However if they are required their occupational exposure is expected to be insignificant (i.e., less than 0.1 man-rem). The existing low density racks will be decontaminated upon removal from the pool, packaged and shipped intact to a disposal site as low concentration radioactive waste. Although this is the present disposal plan of the licensee, because of restrictive limitation of radioactive waste burial, he may choose to consider an alternative option of cutting the racks into smaller sections in order to reduce the volume for burial. Based on relevant experience from other licensees who have performed this operation, the licensee has learned that volume reduction techniques for disposal of spent fuel racks has required an additional collective dose of from 0.1 to 3 man-rem depending upon the reduction method used. Therefore, if he does reconsider his disposal method, the licensee will take into account the additional occupational exposure required for volume reduction (i.e., cutting contaminated racks into smaller sections) taking into account occupational exposure and economic and environmental impact. The occupational exposure expected for the modification of Hatch 1, based on disposal of intact racks, is approximately 14 man-rem.

Very little additional exposure is anticipated for modification of Hatch 2 if the pool remains dry prior to the modification. However, if the pool becomes contaminated before reracking, similar techniques will be performed as stated above for Hatch 1, and an additional 14 man-rem would be received. Thus there is some possibility that 28 man-rem could be received for the SFP modification.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee for dose rates in the spent fuel area from radionuclide concentrations in the SFP water and deposited on the SFP walls. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the additional spent fuel in the pool represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. The small increase in additional exposure will not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable and within the limits of 10 CFR Part 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

2.6 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluation dated June 1978. There will be no change in the waste treatment system or in the conclusions given in Section 11.2 of the evaluation of this system because of the proposed modification.

3.0 Summary

Our evaluation supports the conclusion that the proposed modification to the Hatch 1/2 SFP is acceptable because:

(1) The increase in occupational radiation exposure to individuals due to the storage of additional fuel in the SFP would be negligible.

(2) The potential consequences of the postulated design basis fuel handling accident for the SFP are acceptable.

(3) The likelihood of an accident involving heavy loads in the vicinity of the spent fuel pool is sufficiently small that no additional restrictions on load movement are necessary while our generic review of the issues is under way.

4.0 Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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

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ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 74 TO DPR-57
AND
AMENDMENT NO. 15 TO NPF-5

GEORGIA POWER COMPANY, ET AL.

E. I. HATCH NUCLEAR POWER PLANT UNITS 1 AND 2

DOCKET NO. 50-321 AND 50-366

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1.0 Description of the Proposed Action

By application dated July 9, 1979, Georgia Power Company (the licensee) requested an amendment to Facility Operating License DPR-57, which was issued for Edwin I. Hatch Nuclear Plant Unit No. 1 on August 6, 1974, and Facility Operating License NPF-5, which was issued for Edwin I. Hatch Nuclear Plant Unit No. 2 on June 13, 1978. The proposed amendments would allow an increase in the storage capacity of the Unit No. 1 spent fuel pool from 840 to 3171 fuel assemblies, and the Unit No. 2 spent fuel pool from 1120 to 2755 fuel assemblies. This increase in the capacity would be accomplished by installing storage racks with a center-to-center spacing of approximately 6.5 inches between adjacent vertical cells in place of the existing racks which have approximately 12 inch center-to-center spacing between cells. No changes would be made in the overall pool dimensions or the pool cooling and purification systems.

During a normal refueling, about one fourth of the fuel assemblies are replaced by new fuel. The period between refueling intervals normally varies between twelve and eighteen months depending on plant operating history and the system wide outage schedule.

It is desirable to have enough spent fuel pool storage capacity in reserve to allow for a full core offload. Subsequent to the Unit No. 1 refueling outage in 1979, sufficient reserve for full core offload has not existed in the Unit 1 pool. The licensee proposes to commence the installation of the higher density racks in March, 1980.

Environmental impacts of Units 1 and 2, as designed, were considered in the "Final Environmental Statement for THE EDWIN I. HATCH NUCLEAR PLANT UNIT 1 AND UNIT 2" issued October 1972 by the Directorate of Licensing, U.S. AEC, and in the "Final Environmental Statement related to the operation of EDWIN I. HATCH NUCLEAR PLANT UNIT NO. 2" issued March 1978, by the U. S. Nuclear Regulatory Commission. The purpose of this environmental impact appraisal (EIA) is to determine and evaluate any additional environmental impacts which are attributable to the proposed increase in SFP storage capacity.

2.0 Need for Increased Storage Capacity

According to the licensee's planned refueling schedule, with the present storage rack configuration, full core storage reserve capability will be lost in 1983. This prediction is based on maintaining reserve storage for a single core using the combined storage capacities of both spent fuel pools. This is possible because Unit 1 and Unit 2 share a common refueling floor and a transfer canal which connects the two spent fuel pools. While this capability is not necessary to protect the health and safety of the public, it is desirable to reduce occupational exposures. With the present SFP capacity, the licensee will lose all storage capacity in 1985.

As stated by the licensee, the SFP design was predicted on being able to ship spent fuel offiste for processing after a temporary residence time in the pool for decay of short-lived radioactive fission products. However, spent fuel is not currently being reprocessed on a commercial basis in the United States and storage capacity away from reactor sites is available only on an emergency basis. Additional spent fuel storage capacity is eventually expected to become available at facilities provided by the Department of Energy (DOE); various options are being considered which could result in shipments to such interim facilities in 1984 and to long-term disposition facilities commencing during the 1990-1993 time frame. However, these dates are uncertain since the Congress has not yet authorized or funded these facilities. Furthermore, DOE has stated its intent not to accept spent fuel for interim storage until it has decayed for five years and not to accept it for long-term storage until it has decayed for ten years (so that the fuel can be stored dry without forced-air ventilation). The earliest these conditions can be met by spent fuel discharged from Unit 1 would be in the fall of 1982 for interim storage and the fall of 1987 for long-term storage.

Based on the above information, there is clearly a need for additional onsite spent fuel storage capacity to assure continued operation of Units 1 and 2, with full core off-load capability, after the fall of 1983. The proposed expansion of the total SFP capacity to 6026 assemblies* would provide this capability until the fall of 1997, using annual refueling cycles. If longer refueling cycles (such as 18-months) were adopted after the next cycle for Hatch Units 1 and 2, the present full-core off-load capability would not be extended beyond 1983. Thus, additional storage capacity is needed even if extended refueling cycles are adopted.

3.0 The Facility

Units 1 and 2 each have a boiling water reactor (BWR) with a maximum design power level of 2436 megawatts thermal (MWt). Steam generated in the reactor can be used in turbine-generators to produce up to 786 MWe for Unit 2.

Principal features of the facility which are pertinent to this evaluation are briefly described below for convenience in following the discussion in subsequent sections of this appraisal. More details are presented in the final environmental statements (FES mentioned in section 1 and in the Safety Evaluation Reports (SER) issued by the staff in May, 1973 (Unit 1) and June, 1978 (Unit 2).

* 80 spaces are included in the Unit 2 pool by retaining 4 of the existing storage racks. There are also 10 defective fuel locations in each pool.

3.1 Fuel Inventory

The weight of fuel, as uranium in each reactor is approximately 227,000 pounds. The fuel is contained in long sealed tubes called fuel rods. A cluster of 62 fuel rods arranged in a 8x8 array makes up each of the 560 fuel assemblies in a reactor. (Unit 1 has a mixture of 7x7 and 8x8 arrays.)

The proposed modification of the SFP would not change the quantity of uranium fuel used in the reactor over the anticipated operating life of the facility and would not change the rate at which spent fuel is generated by the facility. The added storage capacity would increase the number of spent fuel assemblies that could be stored in the SFP and the length of time that some of the fuel assemblies could be stored in the pool.

3.2 Purpose of the Spent Fuel Pool

Spent fuel assemblies are intensely radioactive due to their fresh fission product content when initially removed from the core and they have a high thermal output. The SFP was designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. The major portion of decay occurs in the first 150 days following removal from the reactor core. After this period, the spent fuel assemblies may be withdrawn and placed in heavily shielded casks for shipment. Space permitting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling.

3.3 Spent Fuel Pool Cooling and Purification System

The SFP is provided with a cooling system to remove residual heat from the fuel stored in the pool and purification equipment to maintain the quality and clarity of the water in which the fuel assemblies are immersed. The system is discussed in detail in Hatch Unit 1 FSAR Section 10.4 and Hatch Unit 2 FSAR Section 9.1.3; and in Section 9.1.3 of the SER.

The cooling system is designed to maintain the pool water temperature at or below 125°F under normal refueling conditions (with 25% of a core that has an average residual time of four years before being placed in the pool 150 hours after shutdown, plus 25% of a core that has been in storage for one year from a previous refueling operation. Under abnormal conditions, the cooling system is designed to maintain the pool water temperature below 150°F after reaching an equilibrium cycle, with an entire core removed. Two cooling loops are provided for Unit 1 and one cooling loop for Unit 2. Each Unit 1 loop has a full capacity (610 gpm) circulating pump and a heat exchanger designed to remove heat from the pool at a rate of 4.25×10^6 BTU/hour. The Unit 2 loop has a full capacity circulating pump (650) gpm and a heat exchanger designed to remove heat from the pool at a rate of 4.25×10^6 BTU/hour. The three loops are cross-connected for flexibility in the event of a component failure.

In operation, a circulating pump draws water from one end of the pool, circulates it through a heat exchanger and filter/demineralizer and returns it to the other end of the pool. The SFP clean up system consists of a filter vessel, a resin trap, a holding pump, a precoat mixing tank and pump and the required piping, valves and instrumentation. There is also a skimmer system to remove surface dust and debris from the SFP.

3.4 Cooling Water Systems

The heat exchangers in the SFP cooling system discharge the heat from the SFP to the Reactor Building Closed Cooling Water (RBCCW) system which is designed to cool auxiliary equipment located in the reactor building. This system is cooled via heat exchangers by water from the plant service water system which is pumped from the river water intake structure to the plant auxiliary cooling systems and returned to the service water discharge.

Details of the Plant Service Water Systems are discussed in Section 9.2.1 of the SER. During full load operation of Units 1 and 2, a total thermal load of approximately 1.5×10^{11} BTU/hour will be dissipated to the environment. Of this amount, approximately 8.5×10^{6} BTU/hour (about 5.6×10^{3}%) will be contributed by the system under normal operating conditions. If necessary to offload a full core to the SFP, the contribution of the service water system would increase to approximately 10.8×10^{6} BTU/hour for a short time, but the total thermal load dissipated by the plant would diminish to about 7.5×10^{10} BTU/hour as one of the units is shut down. Heat in the service water is normally dissipated by evaporation in the cooling towers to the atmosphere.

3.5 Radioactive Wastes

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the Final Environmental Statement (FES) dated March 1978. There will be no change in the waste treatment system described in Section 3.2.3 of the FES because of the proposed modification.

4.0 Environmental Impacts of the Proposed Action

4.1 Land Use

The external dimensions of the SFP will not change because of the proposed expansion of its storage capacity; therefore, no additional commitment of land is required. The SFP is intended to store spent fuel assemblies under water for a period of time to allow shorter-lived radioactive isotopes to decay and to reduce their thermal heat output. This type of use will remain unchanged by the modification but the additional storage capacity would provide for a total of 23 normal refuelings compared to 9 such refuelings at present. Thus, the proposed modification would result in more efficient use of the land already designed for spent fuel storage.

4.2 Water Use

As indicated in Section 2.2 of the attached Safety Evaluation for the proposed modification, we have verified that the existing SFP cooling system can maintain the same pool water temperatures specified for the original fuel storage configuration. Although the heat to be dissipated would increase somewhat, the amount of makeup water required for pool operation would be essentially the same as that previously considered, since the design temperature limits and rate of water circulation through the pool remain the same.

However, storing additional fuel in the SFP would increase the heat load transferred to the Reactor Building Closed Cooling Water (RBCCW) system and then to the plant service water system by a maximum of 1.275×10^7 BTU/hour. This is less than .01% of the total heat load from both Units, and would be dissipated by evaporation from the cooling towers to the atmosphere with no noticeable effects.

4.3 Nonradiological Effluents

No additional chemicals or biocides are to be used because of the SFP expansion. Therefore, the only nonradiological effluent attributable to the amendment would be the additional heat load of up to 1.275×10^7 BTU/hour dissipated from the plant service water system. This additional heat load is negligible compared to the capability of the Plant Service Water System (1.5×10^{11} BTU/hour.

4.4 Radiological Impacts

4.4.1 Introduction

The potential offsite radiological environmental impacts associated with the expansion of the spent fuel storage capacity were evaluated and determined to be environmentally insignificant as addressed below.

The additional spent fuel which would be stored due to the expansion is the oldest fuel which has not been shipped from the plant. This fuel should have decayed at least three years. During the storage of the spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominately nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to radiation levels to which workers in and near the SFP would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the spent fuel pool water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the spent fuel pool during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800 F. A few weeks after refueling, the spent fuel cools in the spent fuel pool so that the fuel clad temperature is relatively cool, approximately 180 F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months.

Based on operational reports submitted by the licensees or discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) (formerly Midwest Recovery Plant) at Morris, Illinois, or at Nuclear Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite spent fuel pool, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

4.4.2 Effect of Fuel Failure on the SFP

Experience indicates that there is little radionuclide leakage from Zircaloy-clad spent fuel stored in pools for over a decade. Operators at several reactors have discharged, stored, and/or shipped relatively large numbers of Zircaloy-clad fuel elements which developed defects during reactor exposures, e.g., Ginna, Oyster Creek, Nine Mile Point, and Dresden Units Nos. 1 and 2. Based on the operational reports submitted by licensees and discussions with the operators, there has not been any significant leakage of fission products from spent reactor fuel stored in the MO pool or the NFS pool. Several hundred Zircaloy-clad assemblies which developed one or more defects in-reactor are stored in the Morris pool without need for isolation in special cans. Detailed analysis of the radioactivity in the pool water indicates that the defects are not continuing to release significant quantities of radioactivity.

A recent Battelle Northwest Laboratory (BNL) report, "Behavior of Spent Nuclear Fuel in Water Pool Storage: (BNWL-2256 dated September 1977), states that radioactivity concentrations may approach a value up to 0.5 μCi/ml during fuel discharge in the SFP. After the refueling, the SFP ion exchange and filtration units will reduce and maintain the pool water in the range of 10^{-3} to 10^{-4} μCi/ml.

In handling defective fuel, the BNL study found that the vast majority of failed fuel does not require special handling and is stored in the same manner as intact fuel. Two aspects of the defective fuel account for its favorable storage characteristics. First, when a fuel rod perforates in-reactor, the radioactive gas inventory is released to the reactor primary coolant. Therefore, upon discharge, little additional gas release occurs. Only if the failure occurs by mechanical damage in the basin are radioactive gases released in detectable amounts, and this type of damage is extremely rare. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels. The second favorable aspect is the inert character of the uranium oxide pellets in contact with water. This has been determined in laboratory studies and also by casual observations of pellet behavior when broken rods are stored in pools.

4.4.3 Radioactive Material Released to Atmosphere

With respect to gaseous releases, the only significant noble gas isotope attributable to storing additional assemblies for a longer period of time would be Krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no significant release of fission products from defected fuel. However, we have conservatively estimated that an additional 161 curies per year of Krypton-85 may be released from the SFP when the modified pools are completely filled from 1960 to 6026 fuel assemblies. This increase would result in an additional total body dose of less than 0.001 mrem/year to an individual at the site boundary. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. The additional total body dose to the estimated population within a 50-mile radius of the plant is less than 0.0004 man-rem/year. This is small compared to the fluctuations in the annual dose this population would receive from natural background radiation. This exposure represents an increase of much less than 0.1% of the exposure from the plant evaluated in the FES. Thus, we conclude that the proposed modification will not have any significant impact on exposures offsite.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings.

Storing additional spent fuel assemblies in the pool should not increase the bulk water temperature during normal refuelings above the 125 F used as a design condition for the present storage capacity. Therefore, there should not be any significant change in the annual release of tritium or iodine as a result of the proposed modification from that previously evaluated in the FES.

Most airborne releases from the plant result from leakage of reactor coolant which contains tritium and iodine in higher concentrations than the spent fuel pool. Therefore, even if there were a slightly higher evaporation rate from the spent fuel pool, the increase in tritium and iodine released from the plant, as a result of the increase in stored spent fuel, would be small compared to the amount normally released from the plant and that which was previously evaluated in the FES. If levels of radioiodine become too high, the air can be diverted to charcoal filters for the removal of radioiodine before release to the environment. The plant radiological effluent Technical Specifications, which are not being changed by this action, restrict the total releases of gaseous radioactivity from the plant including the SFP.

4.4.4 Radioactivity Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. The amount of radioactivity on the SFP filter-demineralizer might slightly increase due to the additional spent fuel in the pool, but this increase of radioactivity should not be released in liquid effluents from the plant. The plant radiological effluent technical specifications, which are not being changed by this action, restrict the total releases of liquid radioactivity from the plant.

The spent fuel pool has its own filter-demineralizer systems and under normal circumstances the SFP water is not transferred to the liquid radwaste system for processing. Therefore no increase in liquid effluents from the plant is expected as a result of the modification. The fuel pool filter-demineralizer resins are periodically backwashed with water whenever the effluent conductivity exceeds specified limits or the differential pressure across the demineralizer exceeds specified limits. Each backwash cycle generates 2.5 ft^3 of spent resin. Spent demineralizer resins are collected in a spent resin tank and processed for modification as described in Section 4.0.

Leakage from the SFP would be collected in leak collection systems which consist of embedded stainless steel channels behind the stainless steel liner plate. These channels direct the flow to the reactor building floor radwaste drain sumps through the pool leak detection system. The leakage would then be transferred to the liquid radwaste system and processed by the system before any water is discharged from the plant. There have not been signs of leakage from the pool from Unit 1. However should leakage occur it can be detected by several methods (e.g., increase of the make-up water, unusual frequency of operation of the sump pump). Presence of large leaks is annunciated in the control-room by level switches on the sumps.

4.4.5 Solid Radioactive Wastes

The concentration of radionuclides in the pool is controlled by the filter-demineralizer and by decay of short-lived isotopes. The activity is high during refueling operations while reactor coolant water is introduced into the pool and decreases as the pool water is processed through the filter-demineralizer. The increase of radioactivity, if any, should be minor because the additional spent fuel to be stored is relatively cool, thermally, and radionuclides in the fuel will have decayed significantly.

While we believe that there should not be a significant increase in solid radwaste due to the modification, as a conservative estimate, we have assumed that the amount of solid radwaste may be increased by 100 cubic feet a year from the filter-demineralizer. This represents a conservative factor of two increase in the present amount of solid waste from the SFPs for the increase of the spent resins from additional backwash cycles. The annual amount of solid waste shipped from the site was about 18,000 cubic feet for 1975 to 1977. If the storage of additional spent fuel does increase the amount of solid waste from the SFP purification systems by about 100 cubic feet per year, the increase in total waste volume shipped would be about 0.5% and would not have any significant environmental impact.

The present aluminum spent fuel racks, control rod storage racks, safety curtains and seismic restraints to be removed from the Hatch 1 SFP because of the proposed modification are contaminated and will be disposed of as low level solid waste. Because the Hatch 2 SFP is uncontaminated, it is expected that the racks removed from Unit 2 will be stored in a warehouse for future sale or use. The licensee has estimated that about 10,000 cubic feet of solid radwaste will be removed from the plant because of the proposed modification and sent to a licensed burial site. However, with contaminated Hatch 2 SFP racks, this amount of radwaste would be increased to about 20,000 cubic feet. Therefore, the total waste shipped from the plant would be increased by less than 3% over the lifetime of the plant. This should not have a significant environmental impact.

4.4.6 Occupational Radiation Exposures

We have reviewed the licensee's plans for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for the operation is estimated by the licensee to be about 14 man-rem for modificaton of Unit 1. If Unit 2 can be modified while it is uncontaminated, no additional occupational exposure will result. If however, Unit 2 becomes contaminated prior to the modification, then an additional 14 man-rem could result. We consider this to be a reasonable estimate because it is based on dose rate measurements and occupancy factors for individuals performing a specific job during the modification. This operation is expected to be a small fraction of the total man-rem burden from occupational exposure.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee for occupancy times and dose rates in the spent fuel pool area. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

4.4.7 Impacts of Other Pool Modifications

As discussed above, the additional environmental radiological impacts in the vicinity of Hatch 1/2 resulting from the proposed modification are very small fractions (less than 1%) of the impacts evaluated in the Hatch 1/2 FES. These additional impacts are too small to be considered anything but local in character.

Based on the above, we conclude that an SFP modification at any other facility should not significantly contribute to the environmental impact of Hatch 1/2 and that the Hatch 1/2 SFP modification should not contribute significantly to the environmental impact of any other facility.

4.4.8 Impacts on the Community

The new storage racks were fabricated offsite and shipped to the Hatch Plant, where they are stored. Only a few truck or rail shipments would be involved in shipment of these racks and disposal of the present ones. The impacts of dismantling the present racks and installing the new ones will be limited to those normally associated with metal working activities. During fuel handling operations, the impacts will be confined to the refueling floor of the reactor building. Consequently, no significant impact on the community is expected to result from the fuel rack conversion or subsequent operation with increased storage of spent fuel in the SFP.

4.5 Evaluation of Radiological Impact

As discussed above, the proposed modification does not significantly change the radiological impact evaluated in the FES.

5.0 Environmental Impact of Postulated Accidents

Although the new high density racks will accommodate a larger inventory of spent fuel, we have determined that the installation and use of the racks will not change the radiological consequences of a postulated fuel handling accident in the SFP area from those values reported in the FES dated June 13, 1978.

Additionally, the NRC staff has under way a generic review of load handling operations in the vicinity of spent fuel pools to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. Because the main crane meets the requirements of NUREG-0554, we have concluded that the likelihood of a heavy load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of SFP are necessary while our review is under way.

6.0 Alternatives

The staff has considered the following alternatives to the proposed expansion of the SFP storage capacity at Hatch Units 1 and 2: (1) reprocessing the spent fuel; (2) shipment of spent fuel to a separate fuel storage facility; (3) shipment of spent fuel to another reactor site; (4) lengthening the fuel cycles; (5) reduced plant operation; and (6) shutdown of Units 1 and 2. These alternatives are discussed below.

6.1 Reprocessing of Spent Fuel

As discussed earlier, none of the three commercial reprocessing facilities in the United States is currently operating. The General Electric Company's Midwest Fuel Recovery Plant at Morris, Illinois (MO) has not been licensed and Nuclear Fuel Services, Inc. (NFS) informed the Nuclear Regulatory Commission on September 22, 1976, that it was "withdrawing from the nuclear fuel reprocessing business." The NFS facility is on land owned by the State of New York and leased to NFS through 1980. The Allied-General Nuclear Services (AGNS) reprocessing plant at Barnwell, South Carolina, received a construction permit on December 18, 1970. In October 1973, AGNS applied for an operating license for the reprocessing facility; construction of the reprocessing facility is essentially complete but no operating license has been granted. On July 3, 1974, AGNS applied for a materials license to receive and store up to 400 MTU of spent fuel in the onsite storage pool, on which construction has also been completed but hearings with respect to this application have not been held and no license has been granted.

In 1976, Exxon Nuclear Company, Inc. submitted an application for a proposed Nuclear Fuel Recovery and Recycling Center (NFRRC) to be located at Oak Ridge, Tennessee. The plant would include a storage pool that could store up to 7,000 MTU in spent fuel. However, licensing review of this application was discontinued in 1977 as discussed below.

On April 7, 1977, the President issued a statement outlining his policy on continued development of nuclear energy in the U.S. The President stated that: "We will defer indefinitely the commercial reprocessing and recycling of the plutonium produced in the U.S. nuclear power programs. From our own experience, we have concluded that a viable and economic nuclear power program can be sustained without such reprocessing and recycling."

On December 23, 1977, the Nuclear Regulatory Commission terminated the fuel cycle licensing actions involving mixed oxide fuel (GESMO) (Docket No. RM-50-5), the AGNS' Barnwell Nuclear Fuel Plant Separation Facility, Uranium Hexafluoride Facility and Plutonium Product Facility (Docket Nos. 50-332, 70-1327 and 70-1821), the Exxon Nuclear Company, Inc. Nuclear Fuel Recovery and Recycling Fuels Plant (Docket No. 70-1432), and the Nuclear Fuel Services, Inc. West Valley Reprocessing Plant (Docket No. 50-201). The Commission also announced that it would not at this time consider any other applications for commercial facilities for reprocessing spent fuel, fabricating mixed-oxide fuel, and related functions. Consideration of these or comparable facilities has been deferred indefinitely. Accordingly, the Staff considers that shipment of spent fuel to such facilities for reprocessing is not a feasible alternative to the proposed expansion of Hatch SFP storage capacity, especially when considered in the relevant time frame - i.e., 1983 and at least several years thereafter - when the expanded capacity will be needed. Even if the government policy were changed tomorrow to allow reprocessing of spent fuel, the present backlog of spent fuel at various plants and the time it would take to bring adequate reprocessing capacity on line would require that current spent fuel be stored somewhere for up to another 10 years.

6.2 Independent Spent Fuel Storage Facility

An alternative to expansion of onsite spent fuel pool storage is the construction of new "independent spent fuel storage installations" (ISFSI). Such installations could provide storage space in excess of 1,000 MTU of spent fuel. This is far greater than the capacities of onsite storage pools. The fuel storage pools at MO and NFS are functioning as smaller ISFSIs although this was not the original design intent. The license for the GE facility was amended on December 3, 1975 to increase the storage capacity to about 750 MTU; and, as of August 30, 1978, 310 MTU was stored in the pool in the form of 1196 spent fuel assemblies. An application for an 1100 MTU capacity addition is pending and the present schedule calls for completion in 1980 if approved. However, by a motion dated November 8, 1977, General Electric requested the Atomic Safety and Licensing Board to suspend indefinitely further proceedings on this application. This motion was granted.

The staff has discussed the status of storage space at Morris with GE personnel. We were informed that GE is primarily operating the MO facility to store either fuel owned by GE (which had been leased to utilities on an energy basis, or fuel which GE had previously contracted to reprocess. We were also informed that the present GE policy is not to accept spent fuel for storage except fuel for which GE has a previous commitment.* There is no such commitment for Hatch spent fuel. Storage of the Hatch spent fuel at the existing reprocessing facilities is not a viable alternative to the expansion of the Hatch spent fuel pools.

*GE letter to NRC dated May 27, 1977. The licensee had a reprocessing contract which was terminated by GE.

The NFS facility has capacity for about 260 MTU, with approximately 170 MTU presently stored in the pool at West Valley. Although the storage pool is not full, NFS has indicated that it is not accepting additional spent fuel, even from the reactor facilities with which it had reprocessing contracts.

If the receiving and storage station at Barnwell is eventually licensed to accept spent fuel, as discussed in Section 6.1, it would be functioning as an ISFSI until the reprocessing facilities there are licensed to operate. The pool has unused space for about 400 MTU, but AGNS has indicated that it does not wish to operate the storage facility without reprocessing. The cost of shipping assemblies from Hatch to Barnwell has been estimated by the licensee as $1,200 per assembly compared to $2,345 per assembly for the proposed expansion at Hatch. Storage charges at AGNS would be additional.

With respect to construction of new ISFSIs, on October 6, 1978 the NRC proposed a new Part 72 of its regulations specifying procedures and requirements for the issuance of relevant licenses, along with requirements for the siting, design, operation and record keeping activities of the facilities (43 FR 46309). The staff has estimated that at least five years would be required for completion of an ISFSI. This estimate assumes one year for preliminary design; one year for preparation of the license application, environmental report, and licensing review in parallel with one year for detail design; two and one-half years for construction and receipt of an operating license; and one-half year for plant and equipment testing and startup.

Industry proposals for additional independent spent fuel storage facilities are scarce to date. In late 1974, E. R. Johnson Associates, Inc. and Merrill Lynch, Pierce, Fenner and Smith, Inc. issued a series of joint proposals to a number of electric utility companies having nuclear plants in operation or contemplated for operation, offering to provide independent storage services for spent nuclear fuel. A paper on this proposed project was presented at the American Nuclear Society meeting in November 1975 (ANS Transactions, 1975 Winter Meeting, Vol. 22, TANSAO 22-1-836, 1975). In 1974, E. R. Johnson Associates estimated the construction cost would be equivalent to approximately $9,000 per spent fuel assembly.

Several licensees have evaluated construction of an ISFSI and have provided cost estimates. In 1975, Connecticut Yankee, for example, estimated that an independent facility with a storage capacity of 1,000 MTU (BWR and/or PWR assemblies) would cost approximately $54 million and take about 5 years to put into operation. The Commonwealth Edison Company estimated the construction cost of an ISFSI in 1975 at about $10,000 per fuel assembly. To this would be added the costs for maintenance, operation, safeguards, security, interest on investment, overhead, transportation and other costs. These costs are significantly larger than the estimated cost of the increased storage capacity which will be obtained by expending the present reactor pools (approximately $2,345/assembly).

For the long term, the U.S. Department of Energy (DOE) is modifying its program for nuclear waste management to include design and evaluation of a long term repository to provide Government storage of unreprocessed spent fuel rods in a retrievable condition. It is estimated that the long-term storage facility will start accepting commercial spent fuel in the time frame of 1990 to 1993. The criteria for acceptance is that the spent fuel must have decayed a minimum of ten years so it can be stored in dry condition without need for forced air circulation.

As an interim alternative to the long term retrievable storage facility, on October 18, 1977, DOE announced a new "spent nuclear fuel policy." DOE will determine industry interest in providing interim fuel storage services on a contract basis. If adequate private storage services cannot be provided, the Government will provide interim fuel storage facilities. These interim facilities would be designed for storage of the spent fuel under water. DOE, through its Savannah River Operations Office, is preparing a conceptual design for an interim spent fuel storage pool of about 5000 MTU capacity. Congressional authorization has been requested to borrow $300 million for design and construction of this facility.

Based on recent DOE testimony before Congress, it appears that the earliest DOE's interim storage pool would be licensed to accept spent fuel would be about 1984. However, DOE has also stated its intent not to accept any spent fuel that has not decayed for a minimum of five years. Since Hatch spent fuel would thus not be accepted before 1984, the licensee would have to store the spent fuel elsewhere until that time, in order to continue operation with full-core off-load capability after the fall of 1983.

Based on the above information, neither an independent spent fuel storage installation or a Government interim storage facility appears to be a feasible alternative to meet the licensee's needs. The staff does not regard the alternative of storing spent fuel at Morris, West Valley or Barnwell as offering a significant environmental advantage over construction and use of an expanded storage facility at Hatch. The availability of this alternative is speculative and it also would be considerably more expensive. Furthermore, constructing a new ISFSI or a Governmental interim storage facility would clearly have a greater environmental impact than the proposed action. It would require additional land and considerable equipment and structures, whereas installing new racks at Hatch requires only the small amount of material necessary to construct the racks and minor personnel exposure during installation, if the present racks are contaminated prior to their removal.

6.3 Storage at Another Reactor Site

A possibility is to ship the spent fuel from Hatch to the licensee's Vogtle Nuclear Plant (a PWR) Unit 1 which has an expected inservice date of November 1984. This schedule cannot prevent Hatch from losing its full core reserve capacity in 1983; furthermore, the estimated cost would be greater than that of expanding the Hatch pools, as shown below:

1.	Cost of BWR spent fuel storage racks	$1,300/assembly
	Installation (9%)	120
	Contingencies (10%)	130
	Engineering, supervision and overhead (including licensing) (20%)	250
		$1,800/assembly
2.	Cost of transportation (with cask rental)	$1,200/assembly
3.	Total Cost	$3,000/assembly

These costs do not reflect the loss of storage space at Vogtle.

Storage of spent fuel at another reactor facility outside the GPC system would be physically possible but is not considered a realistic alternative. Most operating reactors in the United States are experiencing shortages in spent fuel storage capacity and could not efficiently provide storage space for spent fuel from other plants. According to a survey conducted by the former Energy Research and Development Administration, up to 27 of the operating nuclear power plants will lose the ability to refuel during the period 1977-1986 without additional spent fuel storage pool expansions or access to offsite storage facilities. Thus, the licensee cannot assuredly rely on any other power facility to provide additional storage capability except on a short-term emergency basis. If space were available in another reactor facility, it is unlikely that the cost would be less than storage onsite as proposed.

6.4 Lengthening the Fuel Cycle

Most of the present fuel cycles for light water reactors were based on the premise that spent fuel would be reprocessed and the fissionable material recovered and recycled. With the change in national policy to a "throw- away" cycle, the industry is evaluating higher initial loadings, higher burnups, recycling of low burnup fuel assemblies and extension of periods between refuelings. These types of changes generally are not an immediate alternative. To obtain data to support higher burnups, exposure of experimental fuel in reactors for several years will be necessary. The lead time for design and procurement of core reloads is one to two years. However, in the long run, redesigning the fuel cycle can extend the time between refuelings by 50 to 100%. The number of fuel assemblies that would be replaced during each refueling would increase, but the total number of spent fuel assemblies generated over the lifetime of the facility would be reduced.

In planning fuel cycles, however, there are other factors that have to be taken into consideration other than just minimizing the number of spent fuel assemblies generated. For example, utilities normally try to schedule refuelings during the spring and fall to avoid having the facility shut down during peak load periods. The licensee currently designs annual reload cycles for the units at Hatch Nculear Power Station. To date, three annual reload cycles have been completed at Unit 1 and the first cycle is currently in operation at Unit 2.

Based on studies performed to date, GPC currently considers the initiation of extended cycle design to be economically unattractive for Hatch Units Nos. 1 and 2, particularly since the 1980 reload bundles have already been purchased and they are designed for an annual cycle.

The staff has considered the effects of 18-month reload cycles and concluded that adoption of the 18-month cycles after the next cycle for Hatch Units Nos. 1 and 2 would not extend the present full-core off-load capability beyond 1983. Therefore, this arrangement would not meet the station's need for additional storage capacity in 1984 when storage in DOE interim facilities may become possible.

6.5 Reduced Plant Output

Nuclear plants are usually base-loaded because of their lower costs of generating a unit of electricity compared to other thermal power plants on the system. Therefore, reducing the plant output to reduce spent fuel generation is not an economical use of the resources available. The total production costs remain essentially constant, irrespective of plant output. Consequently, the unit cost of electricity is increased proportionately at a reduced plant output. We note that Hatch Unit 1 has been operating at a cumulative capacity factor of approximately 60% and Hatch Unit 2 about 75%; but Units 1 and 2 would have to operate at about half of this capacity factor to avoid filling the SFP prior to the fall of 1984, when government interim storage facilities, if available, may accept spent fuel from Hatch. If the plant is forced to substantially reduce output because of spent fuel storage restrictions, the licensee would be required to purchase replacement power or operate its higher cost fossil-fired units, if available, without any accompanying environmental advantage. The cost of electricity would therefore be increased without any likely reduction of environmental impact.

6.6 Shutdown of the Facility

Shutdown of Hatch Units 1 and 2 after the SFP is full would result in cessation of approximately 1500 megawatts of electrical production (at full load). The licensee has estimated that replacement energy conservatively would cost $325,000 per day for each unit shut down, based on the average difference in present fuel costs between fossil-fired generation and nuclear generation on its system. At $650,000 per day for the two units, the estimated cost of $10,770,000 for the proposed expansion of the SFP capacity to avoid such a shutdown would be offset in 9 days. While the availability of replacement energy and its cost in the future are uncertain, it is obvious from the above figures that the alternative of shutting down the facility would result in far greater costs than expanding the SFP storage capacity to allow several years of additional operation until other storage or disposal facilities are available.

The need for Hatch Units 1 and 2 was substantiated in previous licensing actions. The staff is not aware of any reason why that need will substantially diminish in the future. Furthermore, since the staff previously concluded that Units 1 and 2 can be operated with only minimal environmental impacts, the operation of other generating facilities to meet load requirements during shutdown of these units would not offer a significant environmental advantage. Therefore, we do not regard shutdown of these units to be a desirable alternative to the proposed action.

6.7 Comparison of Alternatives

In Section 4 the incremental environmental impacts of the proposed expansion of the SFP storage capacity were evaluated and were found to be insignificant. Therefore, none of the alternatives to this action offers a significant environmental advantage. Furthermore, alternatives (1), reprocessing, and (2), storage at an independent spent fuel storage facility, are not presently available to the licensee and are not likely to become available in time to meet the licensee's need. Alternative (3), shipment to another reactor site, would be a short-term solution but would eventually involve shipment to another temporary storage facility. Alternative (4), lengthening the fuel reload cycle would not alleviate the licensee's need for additional storage capacity after 1983. Alternatives (5), reducing the plant output, and (6), shutdown of the facility, would both entail substantial additional expense for replacement electrical energy which may not be available for prolonged periods of time.

Table 1 presents a summarized comparison of the alternatives, in the order presented in subsections 6.1 through 6.6. From inspection of the table, it can be seen that the most cost effective alternative is the proposed spent fuel pool modification, which is included as alternative (7). The SFP modification would provide the required storage capacity, while minimizing environmental effects, capital cost and resources committed. The staff therefore concluded that expansion of the Hatch SFP storage capacity is superior to the alternatives available or likely to become available within the necessary time frame.

TABLE 1

COMPARISON OF ALTERNATIVES

	Alternative	Cost	Benefit
1.	Reprocessing of Spent Fuel	>$10,000/assembly	Continued production of electrical energy by Units 1 & 2. This alternative is not available either now or in the foreseeable future.
2a.	Storage at Reprocessor's Facility	$3,000 to $6,000/assembly per yr* plus shipping costs of $12,000 per assembly.	Continued production of electrical energy by Units 1 & 2. This alternative is not available now or in the foreseeable future.
2b.	Storage at a new Independent Facility	$20,000-$40,000/assembly plus operating and transportation costs, and environmental impacts related to development of a new facility.	Continued production of electrical energy by Units 1 & 2. This alternative could not be available for at least 4 years.
3.	Storage at Other Nuclear Plants	$1,200/assembly for shipment to Vogtle, plus $1,200/assembly for subsequent shipment to an ISFSI; increased environmental costs of extra shipping and handling.	Continued production of electrical energy. However, this alternative is unlikely to be available except at Vogtle, and then only after 1983 or 1984.
4.	Lengthening Fuel Cycle	$1,000 per storage space saved,** plus replacement electricity (see atl. 6).	Continued production of electrical energy by one unit for an additional year.
5.	Reduction in Plant Output	See below for replacement electricity costs. Amount of replacement required would be equivalent to at least 50% reduction in rated output of Units 1 and 2.	Continued production of electrical energy by Units 1 and/or 2 - but at much higher unit cost. The generation of replacement electricity elsewhere would probably create no less impacts.

*Since NFS and MO are not accepting spent fuel for storage, the cost range reflects prices that were quoted in 1972 to 1974. GE estimated that if they were to accept spent fuel on a temporary basis until a utility could locate other storage space, it would probably be at the rate of $30,000 per MTU, which equates to about $5,500 per BWR assembly.

**Based on estimated R&D costs, differential fuel costs and costs for revised ECCS and reload analyses.

	Alternative	Cost	Benefit
6.	Reactor Shutdown	Replacement electricity costs are estimated to be as much as $650,000/day if both units are shut-down, plus the costs of maintenance and security of the plant.	Environmental impacts associated with plant operation would cease but the generation of replacement electricity elsewhere would probably create no less impacts.
7.	Increased storage capacity of Hatch SFP	$7,345/assembly space added	Continued production of electrical energy by Hatch Units 1 & 2

Note: This cost-benefit analysis was commenced prior to the issuance of NUREG-0575, Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel dated August 1979, and is provided in lieu of a reference to the generic statement.

7.0 Evaluation of the Proposed Action

7.1 Unavoidable Adverse Environmental Impacts

7.1.1 Physical Impacts

As discussed in Sections 4.1, 4.2 and 4.3, expansion of the SFP storage capacity would not result in significant adverse environmental impacts on the land, water, air or biota of the area.

7.1.2 Radiological Impacts

As discussed in Section 4.0, expansion of the storage capacity of the SFP will not create any significant additional radiological effects. The additional total body dose that might be received by an individual at the site boundary or the estimated population within a 50-mile radius is less than 0.001 mrem/yr and 0.0004 man-rem/yr, respectively. These exposures are small compared to the fluctuations in the annual dose this population receives from background radiation. The population exposure represents an increase of less than 0.1% of the exposures from the plant evaluated in the FES. The occupational radiation exposure of workers during removal of the present storage racks and installation of the new racks is estimated by the licensee to be about 14 man-rem for the modification of Hatch 1. Modification of Hatch 2 is not expected to provide occupational exposure if the work can be performed during its uncontaminated status. However, if the Hatch 2 SFP becomes contaminated prior to modification, an additional 14 man-rem will result. This is a small fraction of the total man-rem burden from occupational exposure at the plant. Operation of the plant with additional spent fuel in the SFP should add less than one percent to the present total annual occupational exposure at this facility.

7.2 Relationships Between Local Short-Term Use of Man's Environment and the Maintenance and Enhancement of Long-Term Productivity

Expansion of the SFP storage capacity would permit more efficient use of the land already committed to this purpose. There would be no other changes from the evaluation in the FES.

7.3 Irreversible and Irretrievable Commitments of Resources

7.3.1 Water, Land and Air Resources

The proposed action would not result in any significant changes in the commitments of water, land and air resources identified in the FES.

7.3.2 Material Resources

Under the proposed modification, the present spent fuel storage racks would be replaced by higher-density racks that will increase the SFP storage capacity from 1960 to 5926 fuel assemblies. In its submittal, the licensee estimated that approximately 580,000 pounds of type 304 stainless steel will be required. This is a small percentage of the stainless steel used annually in the United States (about 2.8×10^{11} lb) and does not represent a significant commitment of resources. No other material resources will be required since the fuel pool will otherwise remain unchanged.

If the present storage racks are replaced before being contaminated, as expected, they will be scrapped and the materials can be reused.

Longer term storage of spent fuel assemblies withdraws the unburned uranium from the fuel cycle for a longer period of time. Its usefulness as a resource in the future, however, is not changed. The provision of longer onsite storage does not result in any cumulative effects due to plant operation since the throughput of materials does not change. Thus, the same quantity of radioactive material will have been produced when averaged over the life of the plant. This licensing action would not constitute a commitment of resources that would affect the alternatives available to other nuclear power plants or other actions that might be taken by the industry in the future to alleviate fuel storage problems.

7.4 Commission Policy Statement Regarding Spent Fuel Storage

On September 16, 1975, the Commission announced (40FR42801) its intent to prepare a generic environmental impact statement on handling the storage of spent fuel from light water reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement. The Final Statement was issued on August 3, 1979. (Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel" NUREG-0575, August, 1979).

The Commission directed that in the consideration of any such proposed licensing action, among other things, the following five specific factors should be applied, balanced, and weighed in the context of the required environmental statement or appraisal.

1. Is it likely that the licensing action proposed here would have a utility that is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel capacity?

As discussed in this EIA, the Hatch SFP is not expected to have sufficient storage capacity available for off-loading a full-core after the reloads of Units 1 and 2 are accomplished by the end of 1983. Lacking assurance that storage capacity will be available elsewhere except on an emergency basis, expansion of the SFP capacity will therefore be necessary if that capability is to be maintained. It is also doubtful that the licensee could ship spent fuel to interim storage facilities being proposed by DOE prior to November 1984 because of DOR's intent not to accept spent fuel until it has decayed for five years. This is well beyond the end of 1983 when the licensee expects to need space in the SFP in order to accomplish the reloads scheduled for that time. Furthermore, there is a growing need for offsite storage facilities to accommodate spent fuel which has been accumulating at other reactor sites for years. We have therefore concluded that a need for additional SFP storage capacity exists at Hatch which is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity.

2. **Is it likely that the taking of the action here proposed prior to the preparation of the generic statement would constitute a commitment of resources that would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity?**

The only material resources needed for the proposed action would be approximately 580,000 pounds of type 304 stainless steel. This is less than 0.0001 percent of the stainless steel used annually in the United States. The non-material resources required would be primarily the engineering talent and about 5000 man-hours of labor to accomplish the SFP modification.

The increased storage capacity of the Hatch spent fuel pool was also considered as a nonmaterial resource and was evaluated relative to proposed similar licensing actions at other nuclear power plants, fuel reprocessing facilities and fuel storage facilities. We have determined that the proposed expansion in the storage capacity of the SFP is only a measure to allow for continued operation of this facility, and it will not affect similar licensing actions at other nuclear power plants. In 1999, the modified pool is estimated to be full if no fuel is removed. At that time, the licensee will need to ship spent fuel to other storage or disposal facilities which are being contemplated by industry and the Department of Energy. Such facilities will be needed even earlier to accommodate spent fuel from other nuclear power plants.

We have therefore concluded that the expansion of the SFP at Hatch, prior to issuance of the final generic statement, does not constitute a commitment of either material on nonmaterial resources that would tend to significantly foreclose the alternatives available with respect to any other individual licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity.

3. Can the environmental impacts associated with the licensing action here proposed be adequately addressed within the context of the present application without overlooking any cumulative environmental impacts?

Potential nonradiological and radiological impacts resulting from the fuel rack conversion and subsequent operation of the expanded SFP at this facility were considered by the staff.

No environmental impacts on the environs outside of the spent fuel storage building are expected during removal of the existing racks and installation of the new racks. The impacts within this building are expected to be limited to those normally associated with metal working activities and to the occupational radiation exposure to the personnel involved.

The additional thermal effluent from the station and the additional water use associated with storage of the greater number of spent fuel assemblies were determined to be very small compared to those presently associated with Units 1 and 2. Expansion of the SFP would not result in radioactive effluent releases that could significantly affect the quality of the human environment during either normal operation of the expanded SFP or under postulated fuel handling accident conditions.

We have therefore concluded that the environmental impacts associated with this licensing action have been adequately addressed without overlooking any cumulative impacts.

4. Have the technical issues which have arisen during the review of this application been resolved?

This Environmental Impact Appraisal and the related Safety Evaluation adequately address the health, safety and environmental technical issues which have arisen during consideration of this application.

5. Would a deferral or severe restriction on this licensing action result in substantial harm to the public interest?

The staff has evaluated the impact of deferral of the proposed action as it relates to the public interest. We have found that there are significant economic advantages associated with this proposed action, and that expansion of the storage capacity of the SFP will have a negligible environmental impact. Therefore, it is clear that the proposed action itself is in the public interest.

While it is true that Hatch does not face certain shutdown until 1984, there are other factors which weigh in favor of issuing the proposed amendment now. Following the refueling of Unit 2 in the spring of 1983, the existing SFP will not have sufficient room to accommodate a full core (560 assemblies) should this be necessary to effect repairs, for example, to return the unit to service. After this point in time, Hatch faces the possibility of shutdown at any time due to lack of a full core reserve in the SFP. While no serious adverse consequences to the public health and safety or the environment would likely result from this action itself, the reactor shutdown would, of course, remove the unit from service. This, in turn, could adversely affect the licensee's ability to meet electrical energy needs, or force the operation of other plants which are less economical to operate or have greater environmental impact, thereby resulting in substantial harm to the public interest.

Based on the foregoing, we conclude that public interest consideration weighs in favor of taking the proposed action now.

We have applied, balanced, and weighed the five specific factors and have concluded that the proposed expansion of the spent fuel pool is in the public interest.

8.0 Benefit-Cost Balance

As discussed in Section 4 of this assessment, expansion of the Hatch SFP storage capacity would not result in any significant adverse environmental impacts on the land, water, air or biota of the area and it would not create any significant radiological effects.

During construction, the impacts on the community would be limited to those of a few truck or rail shipments carrying the new storage racks to the station and removing the present racks. No incremental occupational exposure of workers would occur if the modification is accomplished, as planned, before the present racks must otherwise be used for storage of spent fuel beginning in the fall of 1980. However, if the racks are removed after being contaminated, the total occupational exposure is estimated to be less than 28 man-rem.

9.0 Basis And Conclusion For Not Preparing an Environmental Impact Statement

We have reviewed this proposed facility modification relative to the requirements set forth in 10 CFR Part 51 and the Council of Environmental Quality's Guidlines, 40 CFR 1500.6, and have applied, weighed, and balanced the five factors specified by the Nuclear Regulatory Commission in 40 FR 42801. We have determined that the proposed license amendment will not significantly affect the quality of the human environment and that there will be no significant environmental impact attributable to the proposed action other than that which has already been predicted and described in the Final Environmental Statement dated October 1972 and the Unit 2 Final Environmental Statement dated March 1978. Therefore, the staff has found that an environmental impact statement need not be prepared, and that pursuant to 10 CFR 51.5(c), the issuance of a negative declaration to this effect is appropriate.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-321 AND 50-366

GEORGIA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE NOS. DPR-57 AND NPF-5 AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 74 to Facility Operating License No. DPR-57 and Amendment No. 15 to Facility Operating License No. NPF-5 to Georgia Power Company, Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensee), which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant Units 1 and 2 (the facility) located in Appling County, Georgia. The amendments are effective as of the date of issuance.

The amendments authorize the installation and use of new high density storage racks for the storage of spent fuel assemblies in the spent fuel storage pool.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Proposed Issuance of Amendments to Facility Operating Licenses in connection with the amendments was published in the FEDERAL REGISTER on August 15, 1979 (44 FR 47820). A petition for leave to intervene was filed by Georgians Against Nuclear Energy (GANE) on September 14, 1979. GANE withdrew its Petition for Leave to Intervene by letter dated November 2, 1979. An Order Dismissing the Proceeding was issued on November 16, 1979.

The Commission has prepared an Environmental Impact Appraisal of the action being authorized and has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the action significantly greater than that which has been predicted and described in the Commission's Final Environmental Statement for the facility dated October 1972 and March 1978.

For further details with respect to this action, see (1) the application for amendments dated July 9, 1979 and supplements thereto dated July 27, September 21, October 29, November 30, December 31, 1979 and February 18, 1980, (2) Amendment No. 74 to License No. DPR-57, (3) Amendment No. 15 to License No. NPF-5, (4) the Commission's concurrently issued Safety Evaluation, and (5) the Commission's concurrently issued Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Appling County Public Library, Parker Street, Baxley, Georgia. A single copy of items (2), (3), (4) and (5) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 21st day of April 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors