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T. Barnhart

Posted
Amdt. 102
to DPR-57

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

December 7, 1984

Docket No. 50-321

Mr. J. T. Beckham, Jr.
Vice President - Nuclear Generation
Georgia Power Company
P. O. Box 4545
Atlanta, Georgia 30302

Dear Mr. Beckham:

The Commission has issued the enclosed Amendment No. 102 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant, Unit No. 1. The amendment changes the Technical Specifications (TSs) in response to your application dated September 17, 1984.

The amendment revises the Technical Specifications (TSs) to allow up to four previously irradiated fuel bundles to be loaded around each Source Range Monitor, delete the description of the control rod material to permit the use of alternative control rod material (hybrid hafnium) in the control rod assemblies, and revise the definition of Core Alteration to clarify that the definition only applies when fuel is in the vessel.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next Monthly Notice.

Sincerely,

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 102
2. Safety Evaluation

cc w/enclosures:
See next page

Hatch 1/2
Georgia Power Company

50-321/366

cc w/enclosure(s):

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UNITED STATES
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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. 102
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 September 17, 1984, 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 102, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 7, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

| Remove | Insert |
|---|---|
| 1.0-2 | 1.0-2 |
| 3.10-2 | 3.10-2 |
| 3.10-7 | 3.10-7 |
| 5.0-1 | 5.0-1 |

C. Core Alteration - Core alteration shall be the addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of core alterations shall not preclude completion of the movement of a component to a safe conservative position.

D. Design Power - Design power refers to the power level at which the reactor is producing 105 percent of reactor vessel rated steam flow. Design power does not necessarily correspond to 105 percent of rated reactor power. The stated design power in megawatts thermal (MWt) is the result of a heat balance for a particular plant design. For Hatch Nuclear Plant Unit 1 the design power is 2537 MWt. Design power is used as an initial condition in transient and accident analyses.

E. Engineered Safety Features - Engineered safety features are those features provided for mitigating the consequences of postulated accidents, including for example containment, emergency core cooling, and standby gas treatment system.

F. Hot Shutdown Condition - Hot shutdown condition means reactor operation with the Mode Switch in the SHUTDOWN position, coolant temperature greater than 212°F, and no core alterations are permitted.

G. Hot Standby Condition - Hot standby condition means reactor operation with the Mode Switch in the START & HOT STANDBY position, coolant temperature greater than 212°F, reactor pressure less than 1045 psig, critical.

H. Immediate - Immediate means that the required action shall be initiated as soon as practicable, considering the safe operation of the Unit and the importance of the required action.

I. Instrument Calibration - An instrument calibration means the adjustment of an instrument output signal so that it corresponds, within acceptable range and accuracy, to a known value(s) of the parameter which the instrument monitors.

J. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

3.10.C Core Monitoring During Core Alterations

1. During normal core alterations, two SRM's shall be operable; one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant, except as specified in 2 and 3 below.

   For an SRM to be considered operable, it shall be inserted to the normal operating level and shall have a minimum of 3 cps with all rods capable of normal insertion fully inserted.

2. Prior to spiral unloading the SRM's shall be proven operable as stated above, however, during spiral unloading the count rate may drop below 3 cps.

3. Prior to spiral reload, up to four (4) fuel assemblies will be loaded into their previous core positions next to each of the 4 SRM's to obtain the required 3 cps. Until these assemblies have been loaded, the 3 cps requirement is not necessary.

D. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 8.5 feet above the top of the active fuel.

E. Control Rod Drive Maintenance

1. Requirements for Withdrawal of 1 or 2 Control Rods

A maximum of two control rods separated by at least two control cells in all directions may be withdrawn or removed from the core for the purpose of performing control rod drive maintenance provided that:

a. The Mode Switch is locked in the REFUEL position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being

4.10.C Core Monitoring During Core Alterations

Prior to making normal alterations to the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response.

Use of special movable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.

Prior to spiral unloading or reloading the SRM's shall be functionally tested. Prior to spiral unloading the SRM's should also be checked for neutron response.

D. Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be checked and recorded daily.

E. Control Rod Drive Maintenance

1. Requirements for Withdrawal of 1 or 2 Control Rods

a. This surveillance requirement is the same as given in 4.10.A.

3.10.A.2 Fuel Grapple Hoist Load Setting Interlocks

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

3. Auxiliary Hoists Load Setting Interlock

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

B. Fuel Loading

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

C. Core Monitoring During Core Alterations

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

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

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

D. Spent Fuel Pool Water Level

The design of the spent fuel storage pool provides a storage location for 3181 fuel assemblies in the reactor building which ensures adequate shielding, cooling, and the reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the active fuel will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the water level to less than 10 feet above the top of the active fuel. Lines extending below this level are equipped with two check valves in series to prevent inadvertent pool drainage.

E. Control Rod Drive Maintenance

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

## 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.

### 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* (See Page 5.0-1a)

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 arrangement 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.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. DPR-57

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

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-321

## 1.0 Introduction

By letter dated September 17, 1984, Georgia Power Company (the licensee) made application to amend the Technical Specifications for the Edwin I. Hatch Nuclear Plant, Unit 1. The proposed changes will: 1) allow up to four bundles to be loaded in their previous positions around a Source Range Monitor in order to produce the required three counts per second; 2) delete the description of control rod material to provide for the use of improved hybrid control rod assemblies; and 3) revise the definition of Core Alteration to clarify that the definition only applies when fuel is in the vessel.

## 2.0 Evaluation

### 2.1 Number of Assemblies Surrounding Source Range Monitors

The Hatch Unit 1 Technical Specifications require a count rate of three counts per second in Source Range Monitor Channels when fuel is being loaded into the core. A spiral loading technique is used at Hatch, i.e., the core is loaded from the center outward in such a way as to preclude a concave configuration. In order to initiate the loading procedure, previously irradiated fuel bundles are placed around each of the four Source Range Monitor detectors to provide a source of neutrons to the detector. Continuous indication of detector operability is thus obtained. The requested change in the Technical Specifications would increase the number of bundles permitted to four (from the current two) in order to allow for potential extended outages.

The Source Range Monitor detectors are loaded, one in each quadrant, at approximately mid-radius. Four unrodded, fresh, high reactivity assemblies have a k-effective value less than 0.95. The four such groups are separated by sufficient water to preclude neutronic coupling. Thus there is no criticality concern associated with the proposed Technical Specification change.

The sensitivity of the detectors to changes in the core multiplication factor will not be changed by the proposed addition of bundles around the monitors. On the basis of the above discussion, we conclude that the proposed change in Technical Specifications is acceptable.

2.2 Use of Hybrid I Control Rods

The description of the control rod assemblies is being revised to permit the replacement of the standard control rod assemblies with the General Electric Hybrid I control Rod (HICR) assemblies. The use of these control rods in BWRs has been reviewed and approved by the NRC staff (Safety Evaluation letter dated August 22, 1983), and we conclude that their use is acceptable in Hatch Unit 1.

The details of the design and materials will not be included in the revised Technical Specifications. Since descriptions of the standard blades exist in the FSAR and of the HICR blades in approved topical report NEDE-22290-A, and the safety design criteria which control rods must meet are contained in the FSAR and in other Technical Specifications, we conclude that this is acceptable.

2.3 Revision of Core Alteration Definition

The definition of Core Alteration is being revised to insert the phrase "with fuel in the vessel" after "... with the vessel head removed...". This change is being made to clarify the definition in order to permit work on the fuel-free core without the presence of a Senior Reactor Operator. The change is consistent with the original intent of the definition and is in agreement with the definition in the BWR Standard Technical Specifications. We find the change to be acceptable.

3.0 Environmental Considerations

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and a change to a surveillance requirement. We have has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 7, 1984

Principal Contributor: W. Brooks