ATTACHMENT 1

NRC DOCKET 50-321
OPERATING LICENSE DPR-57
EDWIN I. HATCH NUCLEAR PLANT UNIT 1
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE

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

Item	Remove Page	Insert Page	Applicable Significant Hazards Evaluation Section
1	3.10-2	3.10-2	1
	3.10-7	3.10-7	1
2	5.0-1	5.0-1	2
3	1.0-2	1.0-2	3

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 moveable, 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 $\pm$ 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 $\pm$ 30 lb load setting of these hoists is adequate to trip the interlock when a fuel bundle is being handled.

B. Fuel Loading

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

C. Core Monitoring During Core Alterations

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

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

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

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.

ATTACHMENT 2

NRC DOCKET 50-321
OPERATING LICENSE DPR-57
EDWIN I. HATCH NUCLEAR PLANT UNIT 1
REQUEST FOR TECHNICAL SPECIFICATION CHANGES

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

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

BASIS:

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

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

The Hatch 1 Technical Specifications would require that the fuel assemblies be loaded into their previous core position next to each of the four SRMs. The loading of the bundles around the SRMs before attaining the 3 cps is permissible because these bundles were in subcritical configuration when they were removed and therefore will remain subcritical when placed back in the previous position. This request is identical to a request by Plant Hatch Unit 2 which was granted as Amendment No. 39.

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

2. Change the description of the control rod assemblies in Section 5.0 (Major Design Features) of the Hatch-1 Technical Specifications to delete references to the specific materials and details of construction of the control blades.

BASIS:

This change is intended to support the use in Hatch-1 of an arbitrary number (up to 137) Type I General Electric Hybrid I Control Rod (HICR) assemblies containing some hafnium as absorber material in place of the boron carbide control rods presently in use. HICRs are intended to be standard replacement control rod assemblies for the General Electric BWR/4 D-lattice operating reactors. The HICRs form, fit and function are identical to that of the blade it replaces. The HICR is designed to increase control rod assembly life and to eliminate cracking of absorber tubes containing boron carbide (B_4C). The essential differences between the HICR and the BWR Z-4 D-lattice control rod assemblies currently in use are:

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

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

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

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

The details of design and materials for the new blades will not be included in the revised text, because those details are unnecessary and inconsistent with other portions of the Design Features Section which do not provide design or materials details. Safety design bases which must be met by control rods are enumerated in the Hatch-1 FSAR, Chapter 3. Analyses documented in the approved topical report, NEDE-22290-A, have shown that those design bases are met by the HICR blades; therefore, use of these blades will cause no reduction in the margin of safety.

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

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

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

As shown above, use of the HICR assemblies in Hatch-1 does not increase the probability or consequences of a previously analyzed accident, nor does it significantly reduce any safety margin. The result of this design change is clearly within all acceptance criteria given in the Hatch-1 FSAR as noted above.

Consequently, this change will not result in a significant hazards consideration. The same change was approved by Amendment No. 39 for Plant Hatch Unit 2.

3. Revise the definition of CORE ALTERATION to clarify that core alterations only occur when there is fuel in the reactor vessel.

BASIS:

This Technical Specifications change is proposed to correct an oversight in the current definition which implies that a core alteration can occur even when no fuel is in the reactor vessel. It is reasonable to only classify as a core alteration movement of equipment within the core shroud while the vessel is fueled. This change will replace the current Unit 1 definition with the same one found in Unit 2 Technical Specifications.

The primary detrimental administrative effect of the current Unit 1 definition results from Specification 6.2.2.e, which requires a Senior Reactor Operator (SRO) any time a core alteration is performed. When fuel is in the vessel, an SRO should be and must be present; however, even when the vessel is completely defueled, as it will be during the upcoming outage, current specifications require the SRO's presence. Approval of this request would free the SRO for other duties following reactor vessel defueling.

Incorporation of this change into Technical Specifications would not affect the plant in any mode other than refueling. Because the change only clarifies the definition and makes it the same as Unit 2 such that a "core alteration" does not occur unless there is a core to alter (fuel in the vessel), Plant safety is in no way jeopardized. Consequently, this change is consistent with Item (i) of the "Examples of Amendments that are Considered Not Likely to Involve Significant Hazards Consideration" listed on page 14870 of the April 6, 1983 issue of the Federal Register and will not result in a signigicant hazards consideration.