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.151
License No. DPR-57

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 1 (the facility) Facility Operating License No. DPR-57 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Autnority of Georgia, and City of Dalton, Georgia, (the licensee) dated October 21, 1987, 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 set forth in 10 CFR Chapter I;

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

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

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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. 151 , 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 and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kahtan N. Jabbour, Acting Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 22, 1988

PD#II-3/DRP-I/II MRood/mac 12/ 8 /87 KNJ	NRR/RSXB WHodges 1 12/ 6 /88	PD#II-3/DRP-I/II LCrocker 12/ 8 /87	OGC-Bethesda MYoung 12/ 12 /87
PD#II-3/DRP-I/II KJabbour, Acting PD 01/21/88	AD/DRP-II GLainas 12/21 /87		

ATTACHMENT TO LICENSE AMENDMENT NO. 151

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 vertical lines indicating the areas of change.

Remove Page	Insert Page
3.10-2	3.10-2
3.10-7	3.10-7
5.0-1	5.0-1

3.10.C. Core Monitoring During Core Alterations

1. During normal core alterations, two SRMs 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 SRMs 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 core positions next to each of the 4 SRMs to obtain the required 3 cps. These assemblies may be any which have been shown to meet the criteria for storage in the spent fuel pool. Until these assemblies have been loaded, the 3 cps requirement is not necessary.

4.10.C. Core Monitoring During Core Alterations

Prior to making normal alterations to the core the SRMs 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 SRMs shall be functionally tested. Prior to spiral unloading the SRMs should also be checked for neutron response.

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.

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 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

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 SRMs 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 SRMs 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 SRMs before attaining the 3 cps is permissible because these bundles form a subcritical configuration.

D. Spent Fuel Pool Water Level

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

E. Control Rod Drive Maintenance

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

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 and shall be limited to those fuel assemblies which have been analyzed with NRC-approved codes and methods and have been shown to comply with all Safety Design Bases in the Final Safety Analysis Report (FSAR).

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 are 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 arrangements of fuel in the spent fuel storage racks and in other credible configurations in the spent fuel pool outside the racks shall be evaluated and shown to have 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.

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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. 89
License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit 2 (the facility) Facility Operating License No. NPF-5 filed by Georgia Power Company, acting for itself, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, (the licensee) dated October 21, 1987, 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 89 , 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 and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kahtan N. Jabbour, Acting Director
Project Directorate II-3
Division of Reactor Projects-I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 22, 1988

PD#II-3/DRP-I/II MRood/mac 12/ 8 /87	NRR/RSXB WHodges 1/ 6 /88	PD#II-3/DRP-I/II LCrocker 12/ 9 /87	OGC-Bethesda MYoung 1/12/87
PD#II-3/DRP-I/II KJabbour, Acting PD 01/ 21 /88	AD/DRP-II GLainas 1/ 21 /87		

ATTACHMENT TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Corresponding overleaf pages are provided to maintain document completeness.

Remove Page	Insert Page
3/4 2-3	3/4 2-3
3/4 2-4j	3/4 2-4j
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-7a	3/4 2-7a
3/4 9-4	3/4 9-4
B 3/4 2-2	B 3/4 2-2
B 3/4 2-4	B 3/4 2-4
B 3/4 9-1	B 3/4 9-1
5-3	5-3

AVERAGE PLANAR LINEAR HEAT GENERATION RATE vs
AVERAGE PLANAR EXPOSURE

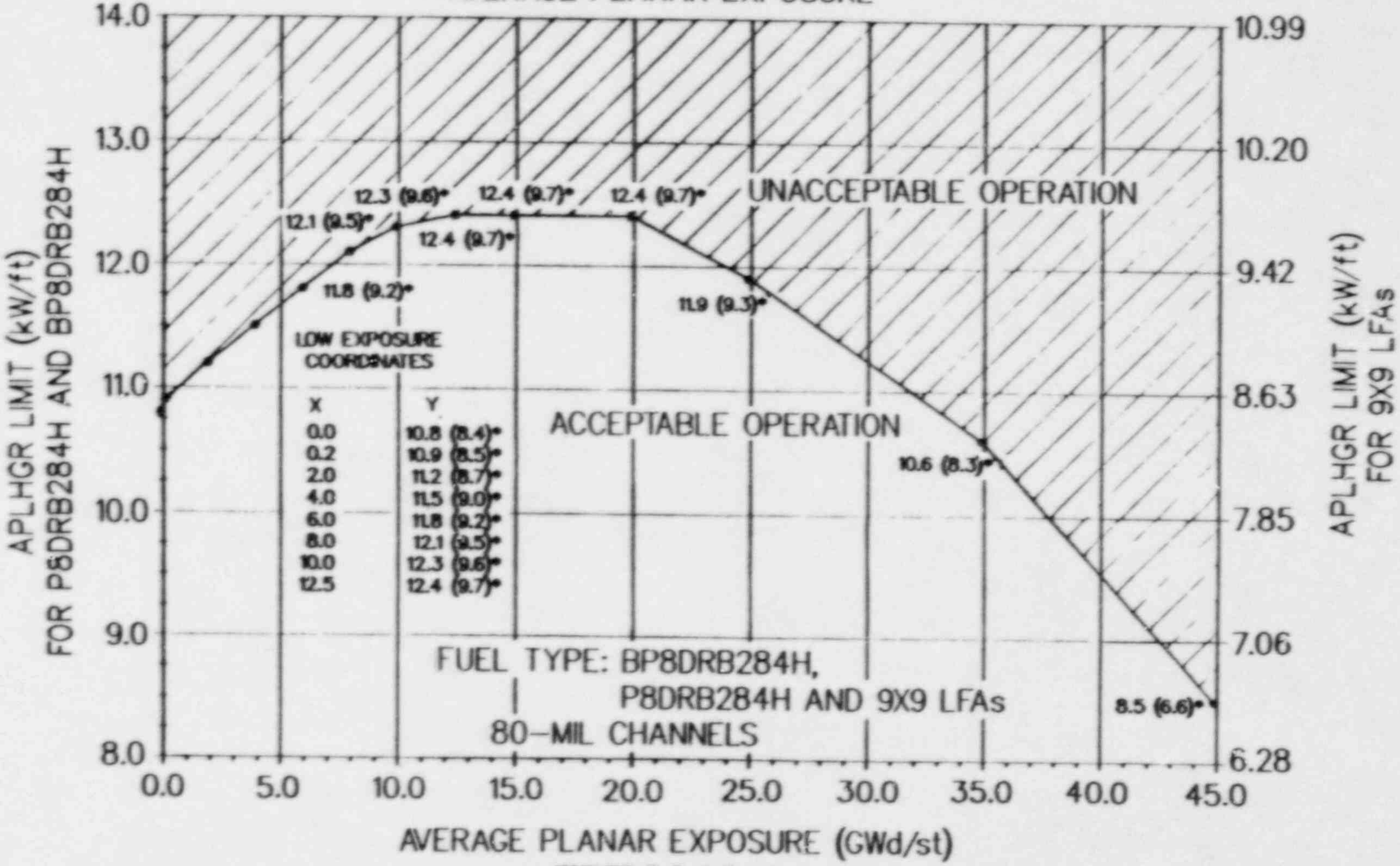
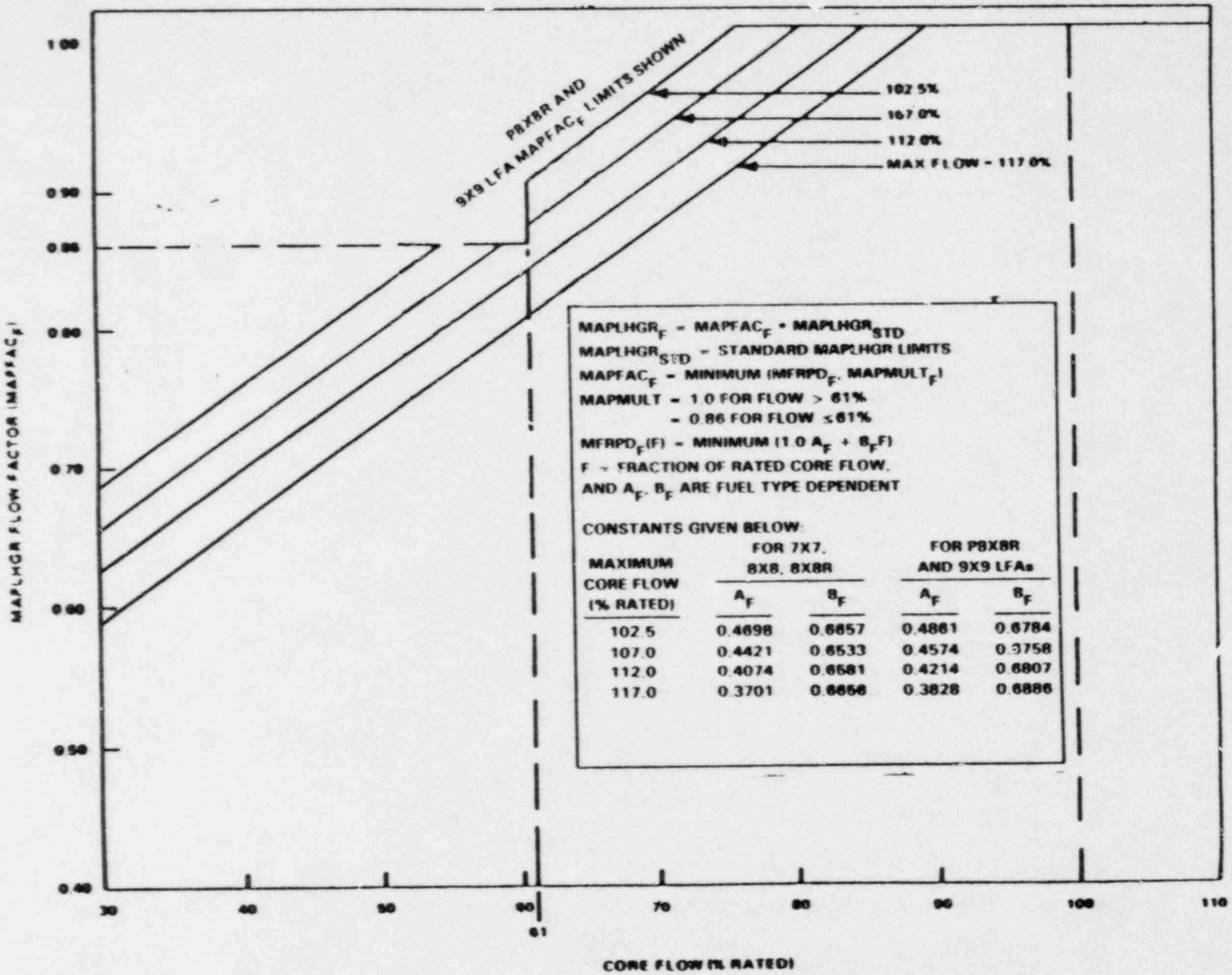


FIGURE 3.2.1-2

*( ) - APLHGR LIMITS FOR 9 X 9 LFAs



$MAPLHGR_F$ = $MAPFAC_F$ • $MAPLHGR_{STD}$
$MAPLHGR_{STD}$ = STANDARD MAPLHGR LIMITS
$MAPFAC_F$ = MINIMUM ($MFRPD_F$, $MAPMULT_F$)
MAPMULT = 1.0 FOR FLOW > 61%
= 0.86 FOR FLOW ≤ 61%
$MFRPD_F(F)$ = MINIMUM (1.0, A_F + $B_F F$)
F = FRACTION OF RATED CORE FLOW,
AND A_F, B_F ARE FUEL TYPE DEPENDENT

CONSTANTS GIVEN BELOW:

MAXIMUM CORE FLOW (% RATED)	FOR 7X7, 8X8, 8X8R		FOR P8X8R AND 9X9 LFAs	
	A_F	B_F	A_F	B_F
102.5	0.4698	0.6657	0.4861	0.6784
107.0	0.4421	0.6533	0.4574	0.3758
112.0	0.4074	0.6581	0.4214	0.6807
117.0	0.3701	0.6656	0.3828	0.6886

FIGURE 3.2.1-12 $MAPFAC_F$

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 ALL MINIMUM CRITICAL POWER RATIOS (MCPRs) for two-loop operation, shall be equal to or greater than the MCPR operating limit (OLMCPR), which is a function of average scram time, core flow, and core power. For 25% ≤ Power < 30%, the OLMCPR is given in Figure 3.2.3-4. For Power ≥ 30%, the OLMCPR is the greater of either:

a. The applicable limit determined from Figure 3.2.3-3, or

b. The appropriate K_p given by Figure 3.2.3-4, multiplied by the appropriate limit from Figure 3.2.3-1 or 3.2.3-2, where τ is the relative measured scram speed with respect to Option A and Option B scram speeds.* If τ is determined to be less than zero, then the OLMCPR is evaluated at $\tau = 0$.

For single-loop operation, the MCPR limit is increased by 0.01 over the comparable two-loop value.

APPLICABILITY: CONDITION 1, when THERMAL POWER ≥ 25% RATED THERMAL POWER

* The specific formula for determining τ is provided in plant procedures.

ACTION:

With MCPR less than the applicable limit determined from Specification 3.2.3.a, or 3.2.3.b for two-loop or single-loop operation, initiate corrective action within 15 minutes and continue corrective action so that MCPR is equal to or greater than the applicable limit within 2 hours or reduce THERMAL POWER to less than or equal to 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 The MCPR limit at rated flow and rated power shall be determined for each type of fuel (8X8R, P8X8R, BP8X8R, 9X9 LFA, and 7X7) from Figures 3.2.3-1 and 3.2.3-2, using:

a. τ = 1.0 prior to the initial scram time measurements for the cycle performed in accordance with Specification 4.1.3.2.a, or

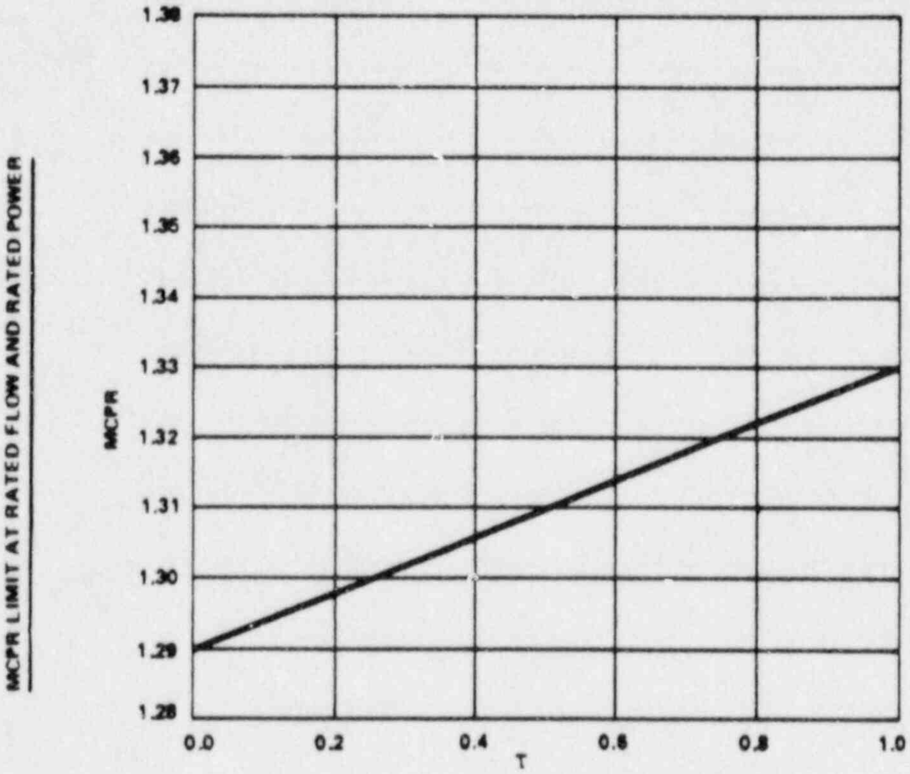
b. τ is determined from scram time measurements performed in accordance with Specification 4.1.3.2. The determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2.

MCPR shall be determined to be equal to or greater than the applicable limit:

a. At least once per 24 hours,

b. Whenever THERMAL POWER has been increased by at least 15% of RATED THERMAL POWER and steady state operating conditions have been established, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.



ALL 8 X 8R AND 9 X 9 LFA FUEL TYPES
FIGURE 3.2.3-1

b. Performance of a CHANNEL FUNCTIONAL TEST:

1. Within 24 hours prior to the start of CORE ALTERATIONS, and

2. At least once per 7 days.

c. Verify that the channel count rate is at least 3 cps at least once per 12 hours during CORE ALTERATIONS, and at least once per 24 hours, except:

1. The 3 cps is not required during core alterations involving only fuel unloading provided the SRMs were confirmed to read at least 3 cps initially and were checked for neutron response.

2. The 3 cps is not required initially on a full core reload. Prior to the reload, up to four fuel assemblies will be loaded into core positions next to each of the 4 SRMs to obtain the required count rate. These assemblies may be any which have been shown to meet the criteria given in Section 5.6.1 of these Technical Specifications for storage in the spent fuel pool.

d. Verifying that the RPS circuitry "shorting links" have been removed and that the RPS circuitry is in a non-coincidence trip mode within 8 hours prior to starting CORE ALTERATIONS or shutdown margin demonstrations.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE

LOSS-OF-COOLANT ACCIDENT ANALYSIS

FOR HATCH-UNIT 2 #

Plant Parameters:

Core Thermal Power .... 2531 Mwt which corresponds to 105% of license core power*

Vessel Steam Output .... 10.96×10^6 lbm/h which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure .... 1055 psia

Design Basis Recirculation Line Break Area For:

a. Large Breaks .... 4.0, 2.4, 2.0, 2.1 and 1.0 ft^2

b. Small Breaks .... 1.0, 0.9, 0.4 and 0.07 ft^2

Fuel Parameters:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.18

A more detailed list of input to each model and its source is presented in subsection 6.3.3 of the FSAR.

For convenience, the APLHGR limits are reported in the units of kW/ft, which is the bundle planar power normalized to the number of fueled rods. Figure 3.2.1-2 shows that the 9x9 LFAs have the same planar power limits as the GE P8DRB284H fuel; however, on a kW/ft basis, the APLHGR limits for the LFAs are 62/79 times the P8DRB284H limits.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification linear heat generation rate limit.

#These are the initial core input parameters. For the updated Loss-of-Coolant Accident Analysis using SAFER/GESTR-LOCA, see Reference 4.

MINIMUM CRITICAL POWER RATIO (Continued)

As depicted on Figure 3.2.3-1 or 3.2.3-2 the 100% power, 100% flow operating limit MCPR (OLMCPR) depends on the average scram time, τ, of the control rods, where:

$$\tau = 0 \text{ or } \frac{\tau_{ave} - \tau_B}{\tau_A - \tau_B}\text{, whichever is greater}$$

where: τ_A = 1.096 sec (Specification 3.1.3.3, scram time limit to notch 36)

$$\tau_B = \mu + 1.65 \left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2} \sigma$$

where: μ = 0.822 sec (mean scram time used in the transient analysis)

σ = .018 sec (standard deviation of μ)

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

where: n = number of surveillance tests performed to date in the cycle

N_i = number of active control rods measured in the ith surveillance test

τ_i = average scram time to notch 36 of all rods in the ith surveillance test

N_1 = total number of active rods measured in 4.1.3.2.a.

The purpose of the $MCPR_f$, and the Kp of Figures 3.2.3-3 and 3.2.3-4, respectively is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power, the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The $MCPR_f$s were calculated such that for the maximum core flow rate and the corresponding THERMAL POWER along the 105% of rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

The core power dependent MCPR operating limit, $MCPR_p$, is the rated power and rated flow MCPR operating limit multiplied by the K_p factor given in Figure 3.2.3-4.

The Kps are established to protect the core from transients other than core flow increases, including the localized event such as rod withdrawal error. The Kps were determined based upon the most limiting transient at the given core power level. For further information on MCPR operating limits for off-rated conditions, see NEDC-30474-P (Reference 2).

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the refuel position ensures that the restrictions on rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage the reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. During the 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 bundles around the SRMs before attaining the 3 cps is permissible because these bundles form a subcritical configuration.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod and prevents two positive reactivity changes from occurring simultaneously.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The reactor building provides secondary containment during normal operation when the drywell is sealed and in service. When the reactor is shutdown or during refueling, the drywell may be open and the reactor building then becomes the primary containment. The refueling floor is maintained under the secondary containment integrity of Hatch-Unit 1.

Establishing and maintaining a vacuum in the building with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches and dampers, is adequate to ensure that there are no violations of the integrity of the secondary containment. Only one closed damper in each penetration line is required to maintain the integrity of the secondary containment.

DESIGN FEATURES

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 137 cruciform-shaped control rod assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 1250 psig, and

c. For a temperature of 575°F

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 17,050 cubic feet at a nominal T_{ave} of 540°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The primary and backup meteorological towers shall be located as shown on Figure 3.11-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with sufficient center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to ≤ 0.95 when flooded with unborated water. The k_{eff} of ≤ 0.95 includes conservative allowances for uncertainties in calculations of both normal and abnormal storage conditions as specified in the FSAR.