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DEC 17 1975

Docket No. 50-321

Georgia Power Company
Oglethorpe Electric Membership Corporation
ATTN: Mr. I. S. Mitchell, III
Vice President and Secretary
Georgia Power Company
Atlanta, Georgia 30302

Gentlemen:

The Commission has issued the enclosed Amendment No. 27 to Facility Operating License No. DPR-57 for Edwin I. Hatch Nuclear Plant Unit 1. This amendment also includes Change No. 26 to the Technical Specifications in accordance with your applications dated July 9, December 2, and December 8, 1975, as supplemented by letters dated August 6, August 29, September 24, October 14, October 21, and December 10, 1975.

The amendment authorizes operation of Hatch Unit 1 (1) with the lower core support bypass flow holes plugged, (2) using operating limits based on the General Electric Thermal Analysis Basis (GETAB), and (3) with modified operating limits based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50 of the Commission's regulations.

In reviewing your applications it was found that certain changes in the proposed Technical Specifications were required. These changes were discussed with and approved by your staff.

The Commission's staff has evaluated the potential for environmental impact associated with operation of Hatch Unit 1 in the manner set forth in item (3) above. From this evaluation, the staff has determined that there will be no change in effluent types or total amounts, no change in authorized power level and no significant environmental impact attributable to that action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal also are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

12/15/75 HKShapar KRG

OFFICE	ORB#3	ORB#3	OELD	ORB#3	RL:AD/ORs	
SURNAME	CParrish:kmf	JGuibert		GLear	KRGoller	
DATE	12/15/75	12/15/75	12/16/75	12/16/75	12/16/75	

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Enclosures:
1. Amendment No. 27
2. Negative Declaration
3. Environmental Impact Appraisal
4. Safety Evaluation
5. Federal Register Notice

cc: G. F. Trowbridge, Esquire
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

12/17/1975

GEORGIA POWER COMPANY

OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
~~License No. DPR-57~~

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

   A. The applications for amendment by Georgia Power Company and Oglethorpe Electric Membership Corporation (the licensees) dated July 9, December 2, and December 8, 1975, and supplements dated August 6, August 29, September 24, October 14, October 21, and December 10, 1975, comply 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility 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, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 26."

Paragraph 2.C.(3) is hereby deleted.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Reactor Licensing

Attachment:
Change No. 26 to the
Technical Specifications

Date of Issuance: December 17, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 27

CHANGE NO. 26 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace page i, page ix, page 1.0-3, page 1.0-6, page 1.0-7, page 1.1-1, pages 1.1-6 thru 1.1-8, pages 1.1-10 thru 1.1-15, pages 3.2-63 thru 3.2-65, page 3.3-5, page 3.3-10, page 3.3-15, page 3.5-9, page 3.5-18, page 3.6-10, pages 3.11-1 thru 3.11-3, Figure 1.1-1, Figure 3.11-1 (Sheet 1) and Figure 3.11-2 with the attached revised pages. No change has been made on page ii, page 1.0-4, page 1.0-5, page 1.1-2, page 1.1-5, Figure 2.1-1, page 3.2-66, page 3.3-4, page 3.3-6, page 3.3-9, page 3.3-16, and page 3.6-9. Delete page 1.1-9, page 1.1-16 and page 1.1-17. Add page 3.11-4, page 3.11.5, Figure 3.11-1 (Sheet 2) and Figure 3.11-3.

TABLE OF CONTENTS

LIST OF FIGURES

K. Instrument Check - An instrument check is the qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

L. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument channel response, alarm and/or initiating action.

M. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the acceptable levels of system performance necessary to assure safe startup and operation of the Unit. When these conditions are met, the Unit can be operated safely and abnormal situations can be safely controlled.

N. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that a Safety Limit will not be exceeded. The region between the Safety Limit and these settings represents margin with normal operation lying on the conservative side of these settings. The margin has been established so that with proper operation of the instrumentation the Safety Limits will never be exceeded.

O. Logic System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure that components are operable per design intent. Where practicable, action will go to completion; e.g., pumps will be started and valves opened.

P. Deleted 26

Q. Operable - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

R. Operating - Operating means that a system or component is performing its intended functions in its required manner.

S. Operating Cycle - An operating cycle is the interval between the end of one scheduled refueling outage and the end of the next subsequent scheduled refueling outage for the same unit.

T. Primary Containment Integrity - Primary containment integrity means that the drywell and suppression chamber are intact and all the following conditions are satisfied:

1. All non-automatic containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed. These valves may be opened to perform operational activities.

2. At least one door in the personnel airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

U. Protective Action - A protective action is an action initiated by the protective system when a limit is reached. A protective action can be at a channel or system level and is essential to the accomplishment of a safety action.

V. Protective Function - A protective function is the monitoring of one or more plant variables or conditions and the associated initiation of intra-system actions which eventually result in protective action.

W. Rated Thermal Power - Rated thermal power means the reactor is operating, at a steady state power of 2436 megawatts thermal. This is also referred to as 100 percent thermal power.

X. Reactor Mode - The reactor mode is established by the Mode Switch position. The switch positions are REFUEL, SHUTDOWN, START & HOT STANDBY and RUN; thus the four possible reactor modes are: Refuel Mode, Shutdown Mode, Start & Hot Standby Mode, and Run Mode.

Y. Reactor Power Operation - Reactor power operation is an operation with the Mode Switch in the START & HOT STANDBY or RUN position with the reactor critical and above 1% of rated thermal power.

Z. Reactor Pressure - Unless otherwise indicated, a reactor pressure listed in these Technical Specifications is that pressure measured at the reactor vessel steam dome.

AA. Refuel Mode - The reactor is in the Refuel Mode when the Mode Switch is in the REFUEL position. When the Mode Switch is in this position, the refueling interlocks are in service.

BB. Refueling Outage - Refueling outage is the period of time between the shutdown of the Unit prior to a refueling and the startup of the Unit after that refueling.

CC. Run Mode - The reactor is in the Run Mode when the Mode Switch is in the RUN position. In this mode the reactor pressure is at or above 880 psig and the reactor protection system is energized with APRM Scram (excluding the APRM 15% of flux scram) and APRM rod blocks in service.

DD. Safety Limit - The Safety Limits are limits below which the reasonable maintenance of the physical barriers which guard against the controlled release of radioactivity is assured. Exceeding such a limit requires Unit shutdown and review by the Atomic Energy Commission before resumption of Unit Operation. Operation beyond such a limit may not in itself result in serious consequences, but it indicates an operational deficiency subject to regulatory review.

EE. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and all the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or are secured in the isolated position.

FF. Shutdown Mode - The reactor is in the Shutdown Mode when the Mode Switch is in the SHUTDOWN position and no core alterations are permitted. When the Mode Switch is placed in the SHUTDOWN position a scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized for two seconds and cannot be reset before ten seconds have elapsed.

GG. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

HH. Start & Hot Standby Mode - The reactor is in the Start & Hot Standby Mode when the Mode Switch is in the START & HOT STANDBY position. When the Mode Switch is in this position, the MSIV closure scram trips are bypassed when the reactor pressure is less than 1045 psig. In this mode the reactor protection system is energized with IRM and APRM (Start & Hot Standby Mode) neutron monitoring system trips and control rod withdrawal inter-locks in service.

II. Surveillance Frequency - Periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus or minus 25%. The operating cycle interval as pertaining to instrument and electrical surveillance shall never exceed 15 months. In the case where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

JJ. Surveillance Requirements - The surveillance requirements are requirements established to ensure that the Limiting Conditions for Operation as stated in Section 3 of these Technical Specifications are met. Surveillance requirements are not required on systems or parts of systems that are not required to be operable or are tripped. If tests are missed on parts not required to be operable or are tripped, then they shall be performed prior to returning the system to an operable status.

KK. Total Peaking Factor (TPF) - The total peaking factor is the highest product of radial, axial, and local peaking factors simultaneously operative at any segment of fuel rod. 26

LL. Transition Boiling - Transition boiling is the boiling that occurs between nucleate and film boiling. Transition boiling is manifested by an unstable fuel cladding surface temperature, rising suddenly as steam blanketing of the heat transfer surface occurs, then dropping as the steam blanket is swept away by the coolant flow, then rising again.

MM. Minimum Critical Power Ratio (MCPR) - Minimum Critical Power Ration (MCPR) is the value of the critical power ratio associated with the most limiting assembly in the reactor core. Critical Power Ratio (CPR) is the ratio of that power in a fuel assembly, which is calculated to cause some point in the assembly to experience boiling transition, to the actual assembly operating power.

NN. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

OO. Unusual Event - An unusal event means the occurrence of:

1. Discovery of any substantial errors in the transient or accident analyses, or in the methods used for such analyses, as described in the FSAR or in the bases for these Technical Specifications.

2. Any substantial variance in an unsafe or less conservative direction from performance specifications contained in these Technical Specifications or from performance specifications relevant to safety related equipment contained in the FSAR. 26

3. Any condition involving a possible single failure which, for a system designed against single failures, could result in a loss of the capability of the system to perform its safety function.

PP. Cumulative Downtime The cumulative downtime for those safety components and systems whose downtime is limited to 7 consecutive days prior to requiring reactor shutdown shall be limited to any 7 days in a consecutive 30 day period.

1.1 FUEL CLADDING INTEGRITY

Applicability

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications

A. Reactor Pressure >800 psia and Core Flow > 10% of Rated.

The existence of a minimum critical power ratio (MCPR) less than 1.06 shall constitute violation of the fuel cladding integrity safety limit.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia)

When the reactor pressure is ≤ 800 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

   a. IRM High High Flux Scram Trip Setting
   The IRM flux scram trip setting shall be ≤ 120/125 of
   26 full scale.

   b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)
   When the Mode Switch is in the REFUEL or START & HOT STANDBY position, the APRM flux scram trip setting shall be ≤ 15/125 of full scale (i.e. ≤ 15% of rated thermal power).

1.1.D. Reactor Water Level (Ho r Cold Shutdown Condition)

Whenever the reactor is in the Hot or Cold Shutdown Condition with irradiated fuel in the reactor vessel, the water level shall be > 25 inches above the top of the active fuel when it is seated in the core.

2.1.A.1.c. APRM H High Flux Scram Trip Setting (Run Mode)

When the Mode Swtich is in the RUN position, the APRM flux scram trip setting shall be:

$$S \leq 0.66\ W + 54\%$$

where:

S = Setting in percent of rated thermal power (2436 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of 2.6, the setting shall be modified as follows:

$$S \leq \{0.66\ W + 54\%\} \frac{2.6}{MTPF}$$

where:

MTPF = The value of the existing maximum total peaking factor

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

Surveillance requirements for MTPF are given in Specification 4.1.B.

2.1.B. Reactor Water Level Trip Settings Which Initiate Core Standby Cooling Systems (CSCS)

Reactor water level trip settings which initiate core standby cooling systems shall be as shown in Tables 3.2-2 thru 3.2-6 at normal operating conditions.

1. HPCI Actuation (LL2)

   HPCI actuation (LL2) shall occur at a water level $\geq$ -38 inches indicated on the Yarway.

2. Core Spray and LPCI Actuation (LL3)

   Core Spray and LPCI actuation (LL3) shall occur at a water level $\geq$ -146.5 inches indicated on the Yarway.

1.1 FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.06. MCPR > 1.06 represents a conservative margin relative to the 26
conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These 26
conditions represent a significant departure from the condition intended by design for planned operation.

A. Reactor Pressure > 800 psia and Core Flow > 10% of Rated.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which could produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum 26
critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables, i.e., normal plant operation presented on Figure 1.1-1 by the nominal expected flow control line. The Safety Limit (MCPR of 1.06) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR >1.32) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit (MCPR of 1.06) is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1.

1.1.A. Reactor Pressure > 800 psia and Core Flow > 10% of Rated (Cont'd)

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.06 would not produce boiling transition.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to HNP-1 operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the boiling transition limit (MCPR = 1.06) operation is constrained to a maximum LHGR of 18.5 Kw/ft. At 100% power this limit is reached with a maximum total peaking factor (MTPF) of 2.6. For the case of the MTPF exceeding 2.6, operation is permitted only at less than 100% of rated thermal power and only with reduced APRM scram settings as required by specification 2.1.A.1.C. 26

B. Core Thermal Power Limit (Reactor Pressure < 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of $28x10^3$ lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than $28x10^3$ lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety system setting will assure that the Safety Limit of 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine

1.1.C. Power Transient (Cont'd)

stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis. 26

D. Reactor Water Level (Hot or Cold Shutdown Condition)

For the fuel in the core during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the fuel during this time, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level became less than two-thirds of the core height. The Safety limit has been established at 25 inches above the top of the irradiated fuel to provide a point which can be monitored and also provide adequate margin.

E. References

1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10958. 26

2.1 FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the HNP-1 Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 2537 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 2436 MWt is the licensed maximum power level of HNP-1, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. Active coolant flow is equal to 88% of total core flow. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted (See Figure 7-1, NEDO-21124 (7))which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the thermal consequences of the transients a MCPR of 1.32 is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady-state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various

2.1 FUEL CLADDING INTEGRITY (Continued)

power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

i. The abnormal operational transients were analyzed to a power level of 2537 MWt.

ii. The licensed maximum power level is 2436 MWt.

iii. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

iv. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher starting power in conjunction with the expected values for the parameters.

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. IRM Flux Scram Trip Setting
The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accomodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded. 26

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Quarter rod density is illustrated in Figure 7.5-8 of the FSAR. Adiitional conserva-

2.1.A.1.a. IRM Flux Scram Trip Setting (Continued)

tism was taken in this analysis by assuming that the IRM channel clos-
est to the withdrawn rod is by-passed. The results of this analysis
show that the reactor is scrammed and peak power limited to one percent
of rated power, thus maintaining MCPR above 1.06. Based on the above 26
analysis, the IRM provides protection against local control rod with-
drawal errors and continuous withdrawal of control rods in sequence
and provides backup protection for the APRM.

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure,
the APRM scram setting of 15 percent of rated power provides adequate
thermal margin between the setpoint and the safety limit, 25 percent
of rated. The margin is adequate to accommodate anticipated maneuvers
associated with power plant startup. Effects of increasing pressure
at zero or low void content are minor, cold water from sources avail-
able during startup is not much colder than that already in the system,
temperature coefficients are small, and control rod patterns are con-
strained to be uniform by operating procedures backed up by the rod
worth minimizer and the Rod Sequence Control System. Worth of indivi-
dual rods is very low in a uniform rod pattern. Thus, of all possible
sources of reactivity input, uniform control rod withdrawal is the most
probable cause of significant power rise. Because the flux distribution
associated with uniform rod withdrawals does not involve high local peaks,
and because several rods must be moved to change power by a significant
percentage of rated power, the rate of power rise is very slow. Gen-
erally, the heat flux is in near equilibrium with the fission rate. In
an assumed uniform rod withdrawal approach to the scram level, the rate
of power rise is no more than 5 percent of rated power per minute, and
the APRM system would be more than adequate to assure a scram before
the power could exceed the safety limit. The 15 percent APRM scram
remains active until the mode switch is placed in the RUN position. 26
This switch occurs when reactor pressure is greater than 880 psig.

c. APRM High High Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated
using heat balance data taken during steady state conditions, reads
in percent of rated thermal power (2436 MWt). Because fission chambers
provide the basic input signals, the APRM system responds directly to
average neutron flux. During transients, the instantaneous rate of
heat transfer from the fuel (reactor thermal power) is less than the
instantaneous neutron flux due to the time constant of the fuel. There-
fore, during abnormal operational transients, the thermal power of the
fuel will be less than that indicated by the neutron flux at the scram
setting. Analyses demonstrate that with a 120 percent scram trip setting,
none of the abnormal operational transients analyzed violate the fuel
Safety Limit and there is a substantial margin from fuel damage. Therefore, 26
the use of flow referenced scram trip provides even additional margin.

2.1.A.1.c. APRM High High Flux Scram Trip Setting (Run Mode) (Continued)

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1.c, when the maximum total peaking factor is greater than 2.6.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.06 when the transient is initiated from MCPR > 1.32.

d. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than 1.06. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore the worst case MCPR which could occur during a steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum total peaking factor exceeds 2.6, thus preserving the APRM rod block safety margin.

2. Reactor Water Low Level Scram Trip Setting (LL1)

The trip setting for low level scram is above the bottom of the separator skirt. This level is > 14.5 feet above the top of the active fuel. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Section 14.3 show that a scram at this level adequately protects the fuel and the pressure barrier. The scram trip setting is approximately 33 inches below the normal operating range and is thus adequate to avoid spurious scrams.

2.1.A.3. Turbine Stop Valve Closure Scram Trip Settings

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of ≤ 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.06 during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure. 26

4. Turbine Control Valve Fast Closure Scram Trip Setting

This turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the turbine bypass. The Reactor Protection System initiates a scram when fast closure of the control valves is initiated by the fast acting solenoid valves. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic control oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts from the one-out-of-two-twice logic input to the reactor protection system. This trip setting, a nominally 50% greater closure time and a different valve characteristic from that of the turbine stop valve, combine to produce transients very similar and no more severe than for the stop valve. This scram is by passed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure. 26

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

The main steam line isolation valve closure scram occurs within 10% of valve movement from the fully open position and thus anticipates the neutron flux and pressure scrams which remain as available backup protection. This scram function is bypassed automatically when the reactor pressure is below 1045 psig and the Mode Switch is not in the RUN position. 26

6. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation of the main steam lines at 880 psig was provided to protect against rapid reactor depressurization and the resulting rapid cooldown of the vessel, which might result from a pressure regulator failure causing inadvertent opening of the control and/or bypass valves.

2.1.A.6. Main Steam Line Isolation Valve Closure on Low Pressure (Continued)

Advantage is taken of the scram trip feature that occurs when the main stream line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity Safety Limit. Operation of the reactor at pressures lower than 880 psig requires that the reactor Mode Switch be in the START & HOT STANDBY position, where protecting of the fuel cladding integrity Safety Limit is provided by IRM's and the APRM 15% scram (Start and Hot Standby Mode). Thus, the combination of main steam line low pressure isolation and isolation valve closure scram trip assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity Safety Limit. The reactor pressure vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the Run Mode is less severe than the loss of feedwater transient. Therefore closure of the MSIV's for thermal transient protection when not in the Run Mode is not 26
required.

7. Main Steam Line Isolation Valve Closure on Low Condenser Vacuum

To provide backup protection for the main condenser against over-pressure due to in-leakage, assuming that the turbine stop valves and bypass valves fail to close, a loss of condenser vacuum initiates automatic closure of all main steam isolation valves, the main steam line drain isolation valve and the reactor water sample line valve (i.e. initiates a Group 1 isolation). Closure of these valves prevents excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system. The low vacuum trip set point is selected far enough above the normal operating vacuum to avoid spurious isolation, however, low enough to provide backup isolation prior to the rupture of the condenser.

B. Reactor Water Level Trip Settings Which Initiate Core Standby Cooling Systems (CSCS)

The core standby cooling systems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each Core Standby Cooling System component was established based on the reactor water low level scram trip setting. To lower the trip setting of the water low level scram trip would increase the capacity requirement for each of the CSCS components. Thus, the reactor vessel water low level scram trip was set low enough to permit margin for operation, yet will not be set lower because of CSCS capacity requirements.

The design of the CSCS components to meet the above guidelines was dependent upon three previously set parameters: the maximum break size, the water low level scram trip setting and the CSCS initiation trip setting. To lower the trip setting for initiation of the CSCS may lead to a decrease in effective core cooling. To raise the CSCS initiation trip setting would be in a safe direction, but it would reduce the margin established to prevent actuation of the CSCS during normal operation or during normally expected transients. Transient and accident analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margin for the fuel.

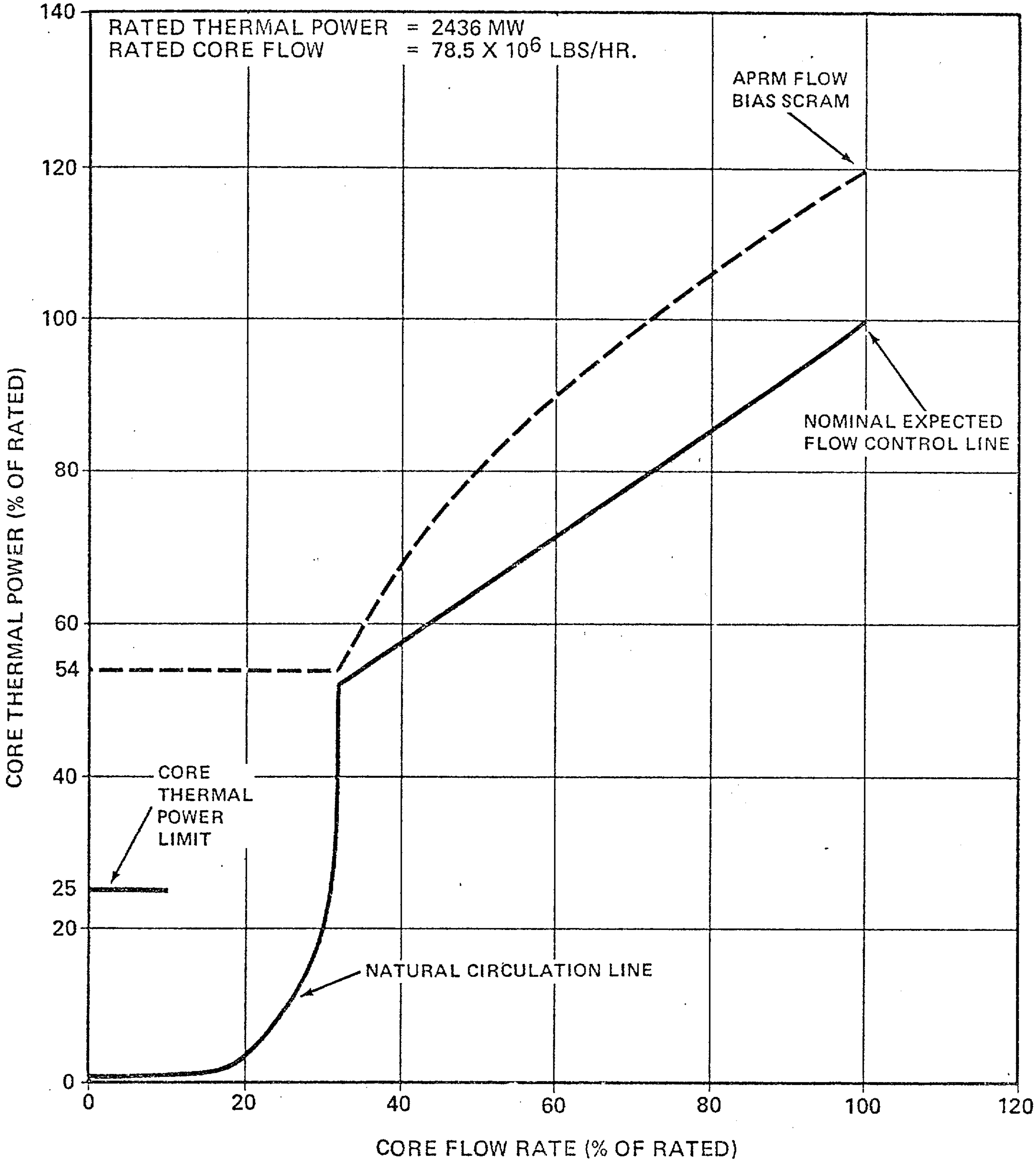

FIGURE 1.1-1
CORE THERMAL POWER SAFETY LIMIT
VERSUS CORE FLOW RATE

WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	READING	INSTRUMENT
(8)	575	+58	YARWAY
(7)	559	+42	GE/MAC
(4)	549	+32	GE/MAC
(3)	529.5	+12.5	BARTON (LL1)
(2)	479	-38	YARWAY (LL2)
(1)	370.5	-146.5	YARWAY (LL3)
(0)	313.5	+313.5	YARWAY

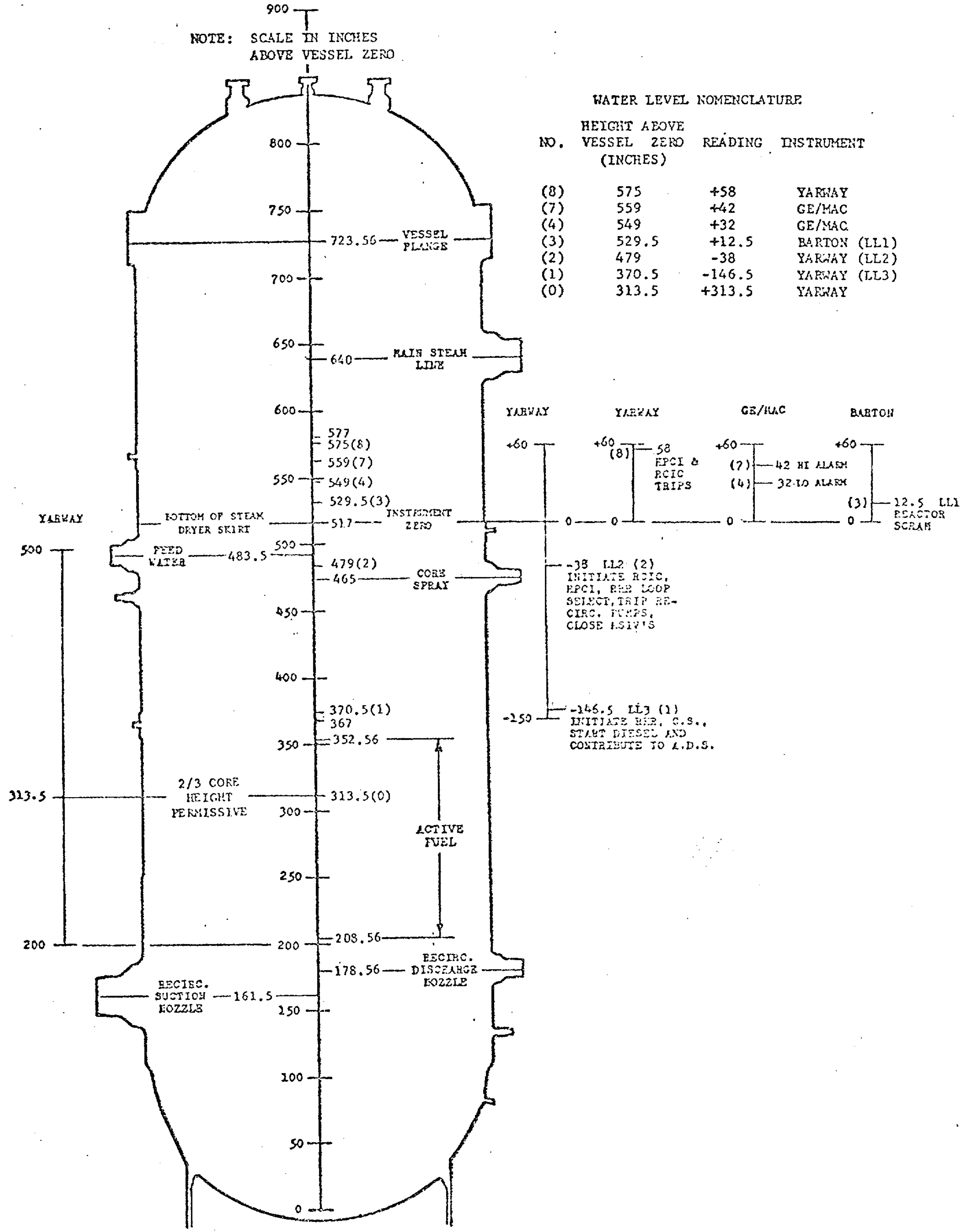

FIGURE 2.1-1 REACTOR VESSEL WATER LEVELS

2.1.C. References

1. FSAR Section 3.7.5.3, Performance Range for Normal Operation.

2. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

3. FSAR Section 3.6.6, Nuclear Evaluations

4. FSAR Section 14.3, Analysis of Abnormal Operational Transients

5. FSAR Section 7.5, Neutron Monitoring System

6. FSAR Section 14, Plant Safety Analysis

7. "Edwin I. Hatch Nuclear Plant Unit 1 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged," NEDO-21124, Nov., 1975. 26

3.2.F.5. Core Spray Pump Discharge Flow

A flow switch is provided downstream of each core spray pump to indicate the condition of each pump. To protect the pumps from overheating at low flow rates a minimum flow bypass line, which routes water from the pump discharge to the suppression chamber, is provided. A single motor-operated valve controls the condition of each bypass line. The minimum flow bypass valve automatically opens upon sensing low flow in the discharge line. The valve automatically closes whenever the flow is above the low flow setting.

6. Core Spray Pump Discharge Interlock

A pressure $\geq$ 100 psig on the core spray pump discharge indicates that the pump has started successfully. The setpoint provides a permissive signal to ADS which allows ADS initiation if other requirements are met.

7. Core Spray Logic Power Failure Monitor

The Core Spray Logic Power Failure Monitor monitors the availability of power to the logic system. In the event of loss of availability of power to the logic system, an alarm is annunciated in the control room.

G. Neutron Monitoring Instrumentation Which Initiates Control Rod Blocks (Table 3.2-7)

These control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to 1.06. The trip logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's or four SRM's will result in a rod block. 26

The minimum instrument channel requirements assure sufficient instrumentation to assure that the single failure criteria is met.

1. SRM

a. Inoperative

This rod block assures that no control rod is withdrawn during low neutron flux level operations unless proper neutron monitoring capability is available, in that all SRM channels are in service or properly bypassed.

b. Not Fully Inserted

Any source range monitor not fully inserted into the core when the SRM count rate level is below the retract permit level will cause a rod block. This assures that no control rod is withdrawn unless all SRM detectors are properly inserted when they must be relied upon to provide the operator with a knowledge of the neutron flux level.

c. Downscale

This rod block assures that no control rod is withdrawn unless the SRM count rate is above the minimum prescribed for low neutron flux level monitoring.

3.2.G.1.d. <u>Upscale</u>

This rod block assures that no control is withdrawn unless the SRM detectors are properly retracted during reactor startup. This setting is selected at the upper end of the range over which the SRM is designed to detect and measure neutron flux.

2. <u>IRM</u>

The trip logic for this function is 1 out of 8; any trip on one of the eight IRM's will result in a rod block. The IRM rod block function provides local as well as gross core protection.

a. <u>Inoperative</u>

This rod block assures that no control rod is withdrawn unless the IRM's are in service.

b. <u>Not Fully Inserted (Refuel and Start & Hot Standby Mode)</u>

This rod block assures that no control rod is withdrawn during low neutron flux level operations unless proper neutron monitoring capability is available in that all IRM detectors are properly located.

c. <u>Downscale</u>

A downscale indication of $\leq$ 5/125 full scale on an IRM is an indication that the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trip is set at $\geq$ 5/125 full scale. This rod block trip is bypassed when the IRM is on the range 1.

d. <u>High Flux</u>

If the IRM channels are in the worst condition of allowed bypass, the scaling arrangement is such that for unbypassed IRM channels a rod block signal is generated before the detected neutron flux has increased by more than a factor of 10.

3. <u>APRM</u>

The trip logic for this function is 1 out of 6; any trip on one of the six APRM's will result in a rod block. The APRM rod block function provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.06 under normal operating conditions. |26

a. <u>Inoperative</u>

This rod block assures that no control rod is withdrawn unless the APRM's are in service.

3.2.G.3.b. Downscale

A downscale indication of $\leq$ 3/125 full scale on an APRM is an indication that the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trip is set $\geq$ 3/125 full scale.

c. 12% Flux (Refuel and Start & Hot Standby Modes)

This rod block anticipates the reactor scram which would occur at 15% rated thermal power (flux), thus preventing the scram by arresting rod movement. Thus the operator is afforded a chance to evaluate the operating conditions and take suitable action before a scram is incurred.

d. High Flux (Flow Referenced)

An APRM rod block trip setting is flow referenced and prevents a significant reduction in MCPR, especially during operation at reduced flow. | 26

4. RBM

The RBM rod block function provides local protection of the core; i.e., the prevention of boiling transition in a local region of the core from a single rod withdrawal error. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only 3% of the operating time in a month and does not significantly increase the risk of an inadvertent control rod withdrawal.

a. Inoperative

This rod block assures that no control rod is withdrawn (above 30% power) unless the RBM channels are in service or are properly bypassed.

b. Downscale

This rod block assures that the RBM's are on scale in the power range or are properly bypassed.

c. High Flux

This rod block prevents the erroneous withdrawal of a single worst case control rod so that local fuel damage does not result. The RBM upscale setting is chosen so that no local fuel damage can occur from a single control rod withdrawal error during power range operation.

H. Radiation Monitoring Systems Which Limit Radioactivity Release (Table 3.2-8)

1. Off-Gas Post Treatment Radiation Monitors

Two air ejector off-gas post treatment radiation monitors are provided in a two from two logic arrangement for the purpose of isolating the off-gas line from the main stack. Each monitor system has three upscale trips at different radiation levels namely HI, HI HI and HI HI HI. Additionally, a downscale trip is provided which results from various inoperative conditions of the monitor channel. Isolation of the off-gas line outlet and drain valves occurs with each monitor indicating HI HI HI, one monitor HI HI HI and the other downscale, or with both monitors downscale. The HI HI HI setpoint corresponds to the instantaneous release limit.

2. Refueling Floor Exhaust Vent Radiation Monitors

Four radiation monitors are provided which initiate isolation of the secondary containment and operation of the standby gas treatment system. The instrument channels monitor the radiation from the refueling area ventilation exhaust ducts.

Two instrument channels with two radiation monitors in each channel are arranged in a two upscale (either channel) trip logic. Trip settings for the monitors in the refueling floor exhaust ventilation ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

3. Reactor Building Exhaust Vent Radiation Monitors

Four radiation monitors are provided which initiate secondary containment isolation, primary containment purge and vent valves isolation and standby gas treatment system actuation. The instrument channels monitor the radiation from the reactor building lower level ventilation exhaust duct.

Two instrument channels with two radiation detectors in each channel are arranged in a two upscale (either channel) trip logic. The trip settings are based on limiting the release of radioactivity via the normal ventilation path and rerouting this activity to be processed through the standby gas treatment system.

4. Control Room Intake Radiation Monitors

Two radiation monitors are provided to initiate isolation of the main control room and recirculation of control room air through filters. The instrument channels monitor radiation from the control room ventilation intake duct.

Two instrument channels are arranged in one upscale, two downscale trip logic. The trip settings are based on limiting the radioactivity from entering the control room from outside.

Chlorine monitor requirements are indicated in Specification 3.12.C.

5. Main Steam Line Radiation Monitors

Although their primary function is to close the MSIV's, the four Main Steam Line radiation monitors also initiate isolation of the mechanical vacuum pump and the gland seal exhauster condenser. The instrument channels monitor the radiation in the main steam line tunnel. The purpose of automatically isolating the mechanical vacuum pump line is to provide timely protection against the release of radioactive materials from the main condenser. Upon receipt of main steam line high radiation signals, the primary containment and reactor vessel isolation control system initiates closure of the mechanical vacuum pump line valve. This isolation precludes or limits the release of fission product radioactivity which, upon fuel failure would be transported from

3.3.C.2 Scram Insertion Times

a. All Operable Control Rods

The average scram insertion time of all operable control rods at a reactor dome pressure $\geq$ 950 psig based on the de-energization of the scram pilot valve solenoids as time zero, shall be no greater than:

%Inserted From Fully Withdrawn	Average Scram Insertion Time (Sec)
5	0.375
20	0.90
50	2.0
90	3.5

26

b. Three Out of Four Rods in a Two-by-Two Array

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array at a reactor dome pressure $\geq$ 950 psig shall be no greater than:

%Inserted From Fully Withdrawn	Average Scram Insertion Time (Sec)
5	0.398
20	0.954
50	2.120
90	3.800

4.3.C.1.b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

4.3.C.2 Scram Insertion Times

a. After each refueling outage all control rods capable of normal insertion shall be scram time tested from the fully withdrawn position after a reactor dome pressure of 950 psig has been attained. This testing must be completed before 40% rated thermal power is exceeded.

b. Routine Time Tests

At 16-week intervals, 10% of the control rods capable of movement with control rod drive pressure shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3.3.C.3 Control Rod Drive Housing Support System

The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A is met.

4.3.C.3. Control Rod Drive Housing Support System

The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

D. Minimum Count Rate for Rod Withdrawal

Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

4.3.D. Minimum Count Rate for Rod Withdrawal

Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observe count rate of at least three counts per second.

E. Rod Worth Inventory Determination

At a specific steady state base condition of the reactor, actual control rod inventory shall be compared to a normalized computed prediction of the inventory. If the difference exceeds 1% Δk, an orderly shutdown shall be initiated and the reactor placed in the Cold Shutdown Condition within 24 hours. The reactor shall remain shutdown until the cause has been determined and corrective actions have been taken as appropriate.

E. Rod Worth Inventory Determination

During the startup test program, and at each startup following refueling outages, the actual rod inventory shall be compared to a normalized computed prediction of the inventory. These comparisions shall be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the actual rod configuation shall be compared to the configuration expected based upon appropriately corrected past data. This comparison shall be made at least every equivalent full power month.

3.3.F. Operation with a Limiting Control Rod Pattern

During operation with a Limiting Control Rod Pattern, either:

1. Both RBM channels shall be operable, or

2. Control rod withdrawal shall be blocked, or

3. The operating power level shall be limited so that the MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod. 26

G. Limiting the Worth of a Control Rod Below 20% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Whenever the reactor is in the Start & Hot Standby or Run Mode below 20% rated thermal power, the Rod Worth Minimizer shall be operable or a second licensed operator shall verify that the operator at the reactor console is following the control rod program.

4.3.F. Operation with a Limiting Control Rod Pattern

During operation when a Limiting Control Rod Pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

G. Limiting the Worth of a Control Rod Below 20% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Prior to the start of control rod withdrawal at startup, and as soon as automatic initiation of the RWM occurs during rod insertion while shutting down, the capability of the Rod Worth Minimizer to properly fulfill its function shall be verified by the following checks.

a. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.

b. The RWM computer on line diagnostic test shall be successfully performed.

c. Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

d. The rod block function of the RWM shall be verified by withdrawing or inserting an out-of-sequence control rod no more than to the block point.

3.3.G.2. Rod Sequence Control System (RSCS)

a. Operability

When the reactor is in the Start and Hot Standby or Run Mode below 20% rated thermal power, the Rod Sequence Control System shall be operable except initially when performing the RWM surveillance tests.

b. Failed Position Switch

Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System if the actual rod position is known. These rods shall be moved in sequence to their correct positions (full in on insertion or full out on withdrawal).

2. Rod Sequence Control System (RSCS)

a. Operability

Prior to the start of control rod withdrawal at startup and as soon as automatic initiation of the RSCS occurs during rod insertion while shutting down, the capability of the Rod Sequence Control System to properly fulfill its function shall be verified by attempting to select and move a rod in each of the out-of-sequence groups.

After 50% of the control rods have been withdrawn and as soon as automatic initiation of the RSCS occurs during rod insertion while shutting down, the operability of the notching restriction shall be demonstrated by attempting to move a control rod more than one notch in the first programmed rod group.

b. Failed Position Switch

A second licensed operator shall verify the conformance to Specification 3.3.G.2.b before a rod may be bypassed in the Rod Sequence Control System.

B. Control Rods

Limiting Conditions for Operation:

Specification 3.3.B.1 requires that a rod which cannot be moved with drive pressure be taken out of service by being disarmed electrically. To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive. Electrical disarming does not eliminate position indication. If the rod is fully inserted and disarmed electrically, it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A. This assures that the core can be shutdown at all times with the remaining control rods assuming the highest worth operable control rod does not insert. An allowable pattern for control rods disarmed electrically, which shall meet this Specification, will be determined and made available to the operator.

Surveillance Requirements:

The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram, since, if a rod can be moved with drive pressure, it will scram because of higher pressure applied during scram. The frequency of exercising the control rods under the conditions of three or more inoperable rods provides even further assurance of the reliability of the remaining control rods. The checks are done at power levels greater than 30% rated thermal power to clear the RWM and RSCS interlocks.

3.3.C. Control Rod Drive System

1. Control Rod Drive Coupling Integrity

Limiting Conditions for Operation:

Operability of the control rod drive system requires that the drive be coupled to the control rod. In the analysis of control rod drop accidents (FSAR subsection 14.4.3), it has been assumed that one control rod drive coupling has lost its integrity. To assure that not more than one coupling could be in this condition, it is required that either a drive is coupled to the control rod or the drive is fully inserted and disarmed electrically. This requirement serves to maintain operation within the envelope of conditions considered by the plant safety analyses.

Surveillance Requirements:

Observation of a response from the nuclear instrumentation during an attempt to withdraw a control rod provides an indication that the rod is following the drive. The overtravel position feature provides a positive check on the coupling integrity, for only an uncoupled drive can reach the overtravel position.

2. Scram Insertion Times

Limiting Conditions for Operation:

The control rod drive system is designed to bring the reactor subcritical at a rate fast enough to prevent excessive fuel damage. The limiting power transient is that resulting from a loss of condenser vacuum (FSAR Appendix G, Event 12, turbine stop-valve closure with closure of the turbine bypass system). Analysis of the transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specification provide the required protection and MCPR remains greater than 1.06. The limit on the number and pattern of rods permitted to have long scram times is specified to assure that the effect of rods of long scram times are minimized in regard to reactivity insertion rate. Grouping of long scram time rods is prevented by not permitting more than one slow rod in any four rod array. The minimum amount of reactivity to be inserted during a scram is controlled by permitting no operable control rod to have a scram insertion time for 90% insertion greater than 7 seconds.

3.3.F. Operation with a Limiting Control Rod Pattern

Surveillance Requirements:

A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit; i.e., operating on a limiting value for APLHGR, LHGR, or MCPR as defined in Specifications 3.11.A, B, and C. During use 26
of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is normally the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

G. Limiting the Worth of a Control Rod Below 20% Rated Thermal Power

1. Rod Worth Minimizer (RWM)

Limiting Conditions for Operation:

The Rod Worth Minimizer (RWM) and the Rod Sequence Control System (RSCS) restrict withdrawals and insertions of control rods to pre-specified sequences. All patterns associated with these sequences have the characteristic that, assuming the worst single deviation from the sequence, the drop of any control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in any pellet average enthalpy in excess of 280 calories per gram. An enthalpy of 280 calories per gram is well below the level at which rapid fuel dispersal could occur (i.e., 425 calories per gram). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Reference Sections 3.6.5.4, 3.6.6, 7.14.5.3, 14.4.2, and Appendix P of the FSAR and NEDO-10527 and supplement thereto.

3.3.G.1. Rod Worth Minimizer (RWM) (Continued)

In performing the function described above, the RWM and RSCS are not required to impose any restrictions at core power levels in excess of 20% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 20%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

At power levels below 20% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calorie per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod sequences and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator or other qualified technical plant employee whose qualifications have been reviewed by the AEC can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural controls to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20%, these devices force adherence to acceptable rod patterns. Above 20% of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20% of rated power are imposed by power distribution requirements as defined in Section 3.11 and 4.11 of these Technical Specifications. Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of $\pm$ 10% of full power the nominal instrument setting is 30% of rated power. Power level for automatic cutout of the RWM function is sensed by feedwater and steam flow and is set nominally at 30% of rated power to be consistent with the RSCS setting.

Surveillance Requirements:

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 20% rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

2. Rod Sequence Control System (RSCS)

a. Operability

Limiting Conditions for Operation:

See bases for Technical Specification 3.3.G.1 Rod Worth Minimizer.

3.5.F Automatic Depressurization System (ADS)

1. Normal System Availability

The seven valves of the Automatic Depressurization System shall be operable: 26

a. Prior to reactor startup from a cold shutdown, or

b. When there is irradiated fuel in the reactor vessel and the reactor is above 113 psig except as stated in Specification 3.5.F.2.

2. Operation with Inoperable Components

If one of the seven ADS valves is 26
known to be incapable of automatic operation, the reactor may remain in operation for a
period not to exceed seven (7) 26
days, provided the HPCI system is operable. (Note that the pressure releif function of these valves is assured by Specification 3.6.H; Specification 3.5.F only applies to the ADS function).

3. Shutdown Requirements

If Specification 3.5.F.1 or 3.5.F.2 cannot be met, an orderly shutdown will be initiated and the reactor pressure shall be reduced to 113 psig or less within 24 hours.

4.5.F Automatic Depressurization System (ADS)

1. Normal Operational Tests

A simulated automatic actuation test shall be performed on the ADS prior to startup after each refueling outage. Surveillance of all relief valves is covered in Specification 4.6.H.

2. Surveillance with Inoperable Components

When it is determined that one 26
of the seven ADS valves is incapable of automatic operation, the HPCI system and the actuation logic of the other ADS valves shall be demonstrated to be operable
immediately and daily 26
thereafter until all seven ADS valves are capable of automatic operation.

F.1. Normal System Availability (continued)

Specification 3.6. states the requirements for the pressure relief function of the valves. It is possible for any number of the valves assigned to the ADS to be incapable of performing their ADS functions because of instrumentation facilures yet be fully capable of performing their pressure relief function.

Because the automatic depressurization system does not provide makeup to the reactor primary vessel, no credit is taken for the steam cooling of the core caused by the system actuation to provide further conservatism to the Core Standby Cooling Systems.

26

2. Operation with Inoperable Components

With one ADS valve known to be incapable of automatic operation six valves remain operable to perform their ADS function. However, since the ECCS Loss of Coolant Accident analysis for small line breaks assumed that all seven ADS valves were operable, reactor operation with one ADS valve inoperable is only allowed to continue for seven (7) days provided that the HPCI system is demonstrated to be operable and that the actuation logic for the (remaining) six ADS valves is demonstrated to be operable. 26

3. Minimum Core and Containment Cooling Systems Availability

The purpose of this Specification is to assure that adequate core cooling equipment is available at all times. If, for example, one core spray loop were out of service and the diesel which powered the opposite core spray were out of service, only 2 RHR pumps would be available. Specification 3.9 must also be consulted to determine other requirements for the diesel generators. In addition, refer to definition 1.0.00 for Cumulative Downtime requirements.

This specification establishes conditions for the performance of major maintenance, such as draining of the suppression pool. The availability of the shutdown cooling subsystem of the RHR system and the RHR service water system ensure adequate supplies of reactor cooling and emergency makeup water when the reactor is in the Cold Shutdown condition. In addition this specification provides that, should major maintenance be performed, no work will be performed which could lead to draining the water from the reactor vessel.

3.6.H. Relief/Safety Valves

When more than one relief/safety valve is known to be failed an orderly shutdown shall be initiated and the reactor depressurized to less than 113 psig within 24 hours. Prior to reactor startup from a cold condition all relief/safety valves shall be operable

4.6.H. Relief/Safety Valves

1. End of Operating Cycle

Approximately one-half of all relief/safety valves shall be benchchecked or replaced with a benchchecked valve each refueling outage. All 11 valves will have been checked or replaced upon the completion of every second operating cycle.

2. Each Operating Cycle

Once during each operating cycle, at a reactor pressure >100 psig each relief valve shall be manually opened until thermocouples downstream of the valve indicate steam is flowing from the valve.

3. Integrity of Relief Valve Bellows

The integrity of the relief valve bellows shall be continuously monitored and the pressure switch calibrated once per operating cycle and the accumulators and air piping shall be inspected for leakage once per operating cycle.

4. Relief Valve Maintenance

At least one relief valve shall be disassembled and inspected each operating cycle.

I. Jet Pumps

Whenever the reactor is in the Start & Hot Standby or Run Mode with both recirculating pumps operating, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

I. Jet Pumps

Whenever both recirculating pumps are operating with the reactor in the Start & Hot Standby or Run Mode, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously.

1. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed.

4.6.I. Jet Pumps (Continued)

2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

3. The diffuser to lower plenum differential pressure reading on an individual jet pump vary from the mean of all jet pump differential pressures by more than 10%.

3.6.J. Recirculation Pump Speeds

1. Two Pump Operation

Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when the power level is greater than 80% and within 15% of each other when the power level is less than 80%.

2. Operation with a single recirculation pump is permitted for 24 hours unless the recirculation pump is sooner made operable. If the pump cannot be made operable, the reactor shall be in cold shutdown within 24 hours. 26

3. Post One Pump Operation

Following one pump operation the discharge valve of the low speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.

K. Structural Integrity of Primary System Boundary

The structural integrity of the primary system boundary shall be maintained at the level required to assure safe operation throughout the life of the unit. The reactor shall be maintained in a Cold Shutdown Condition until each indication of a defect has been investigated and evaluated.

4.6.J. Recirculation Pump Speeds

Recirculation pump speeds shall be recorded at least once per day.

K. Structural Integrity of Primary System Boundary

A preservice inspection of accessible components listed in Table 4.6-1 will be conducted before initial fuel loading to establish a preservice base for later inspections. The nondestructive inspections listed in Table 4.6-1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

3.11 FUEL RODS

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.11-1. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. 26 26

B. Linear Heat Generation Rate (LHGR)

During power operation, the LHGR as a function of core height shall not exceed the limiting value shown in Figure 3.11-2. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the 26 26 26

4.11 FUEL RODS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.11.C. Minimum Critical Power Ratio (MCPR)

During power operation, MCPR shall be $\geq$ 1.32 at rated power and flow. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. For core flows other than rated the MCPR shall be 1.32 times K_f where K_f is as shown in Figure 3.11-3. 26

D. Reporting Requirements

If any of the limiting values identified in Specifications 3.11.A, B. or C. are exceeded, a Reportable Occurrence report shall be submitted.

If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this specification.

4.11.C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.F. 26

3.11 FUEL RODS

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K, even considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50, Appendix K limit. The limiting value for APLHGR is shown in Figures 3.11-1, sheets 1 and 2. 26

3.11.B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at the 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluations below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there 26
have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

D. Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e. there is no allowable time in which the plant can knowingly exceed the limiting values for APLHGR, LHGR, and MCPR. It is a requirement, as stated in Specifications 3.11.A, B, and C that if at any time during steady state power operation, it is determined that the limiting values for APLHGR, LHGR, or MCPR are exceeded, action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving operation beyond a specified limit shall be reported as a Reportable Occurrence. If the specified corrective action described in the LCO's was taken, a thirty-day written report is acceptable.

3.11.E References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August, 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 16, 1974 (USA Regulatory Staff). 26

3. Communication: V.A. Moore to I.S. Mitchell "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

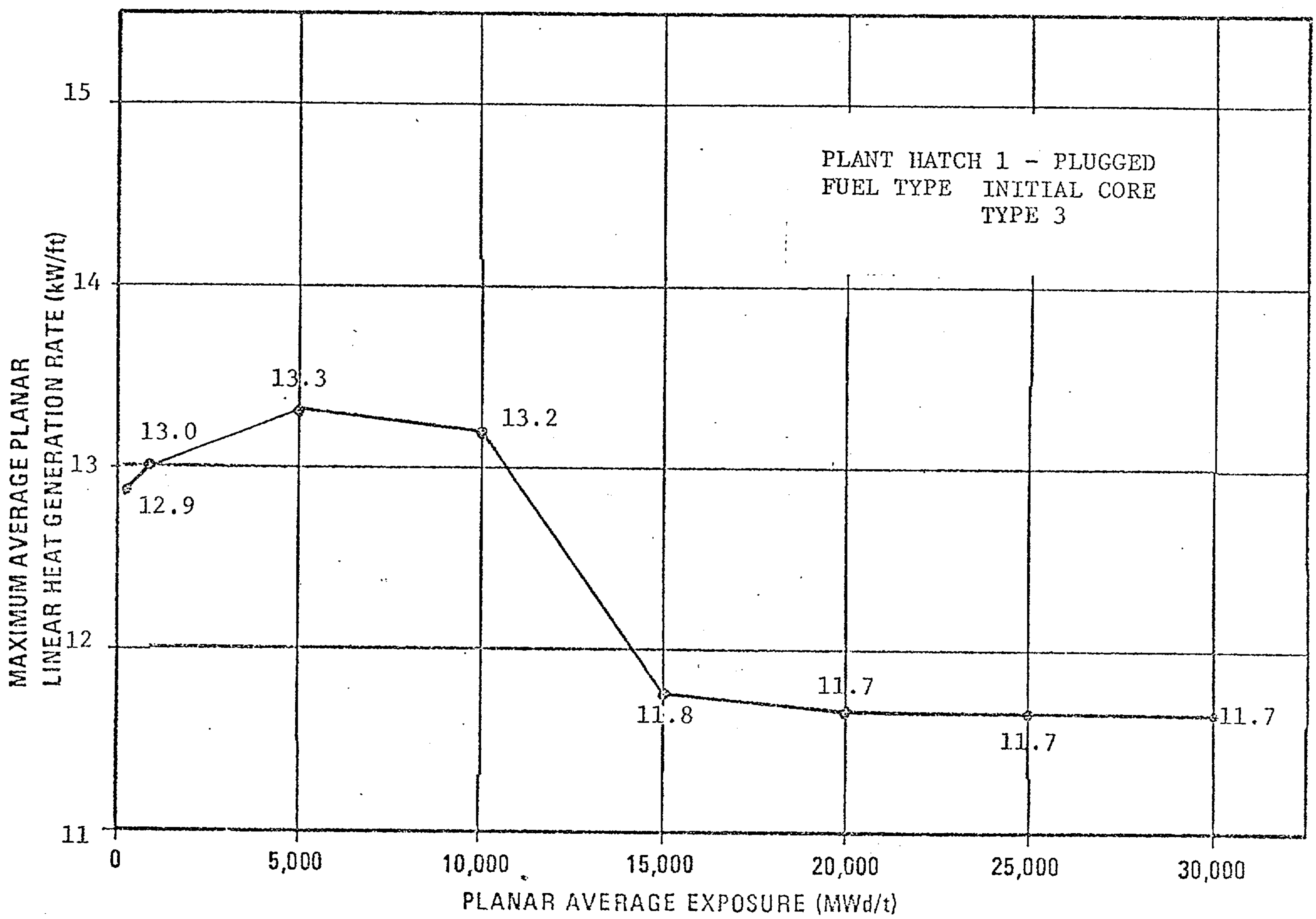

FIGURE 3.11-1 (SHEET 1)
LIMITING VALUE FOR APLHGR
(FUEL TYPE 3)

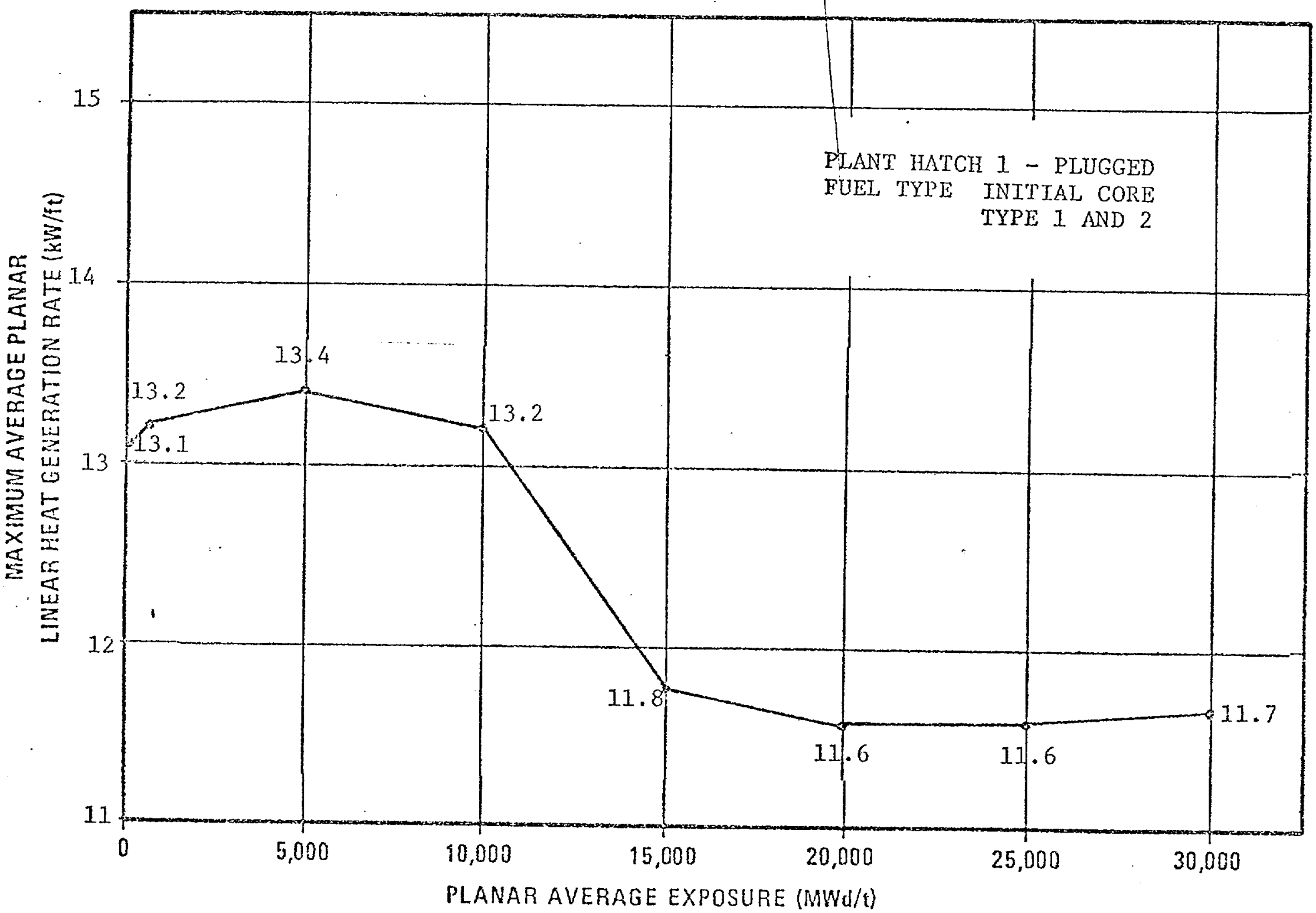

FIGURE 3.11-1 (SHEET 2)
LIMITING VALUE FOR APLHGR
(FUEL TYPES 1 & 2)

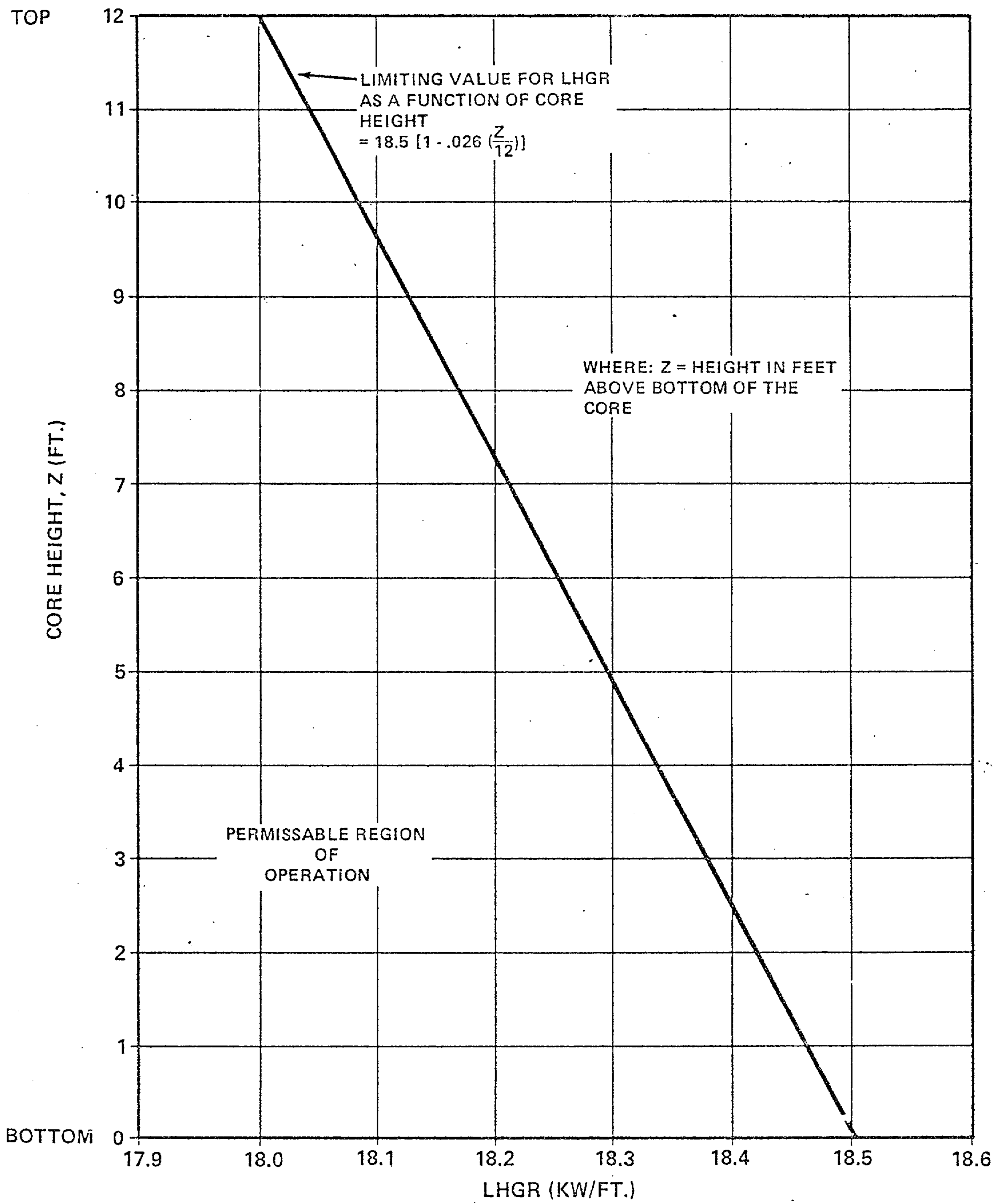

FIGURE 3.11-2

LIMITING VALUE FOR LHGR

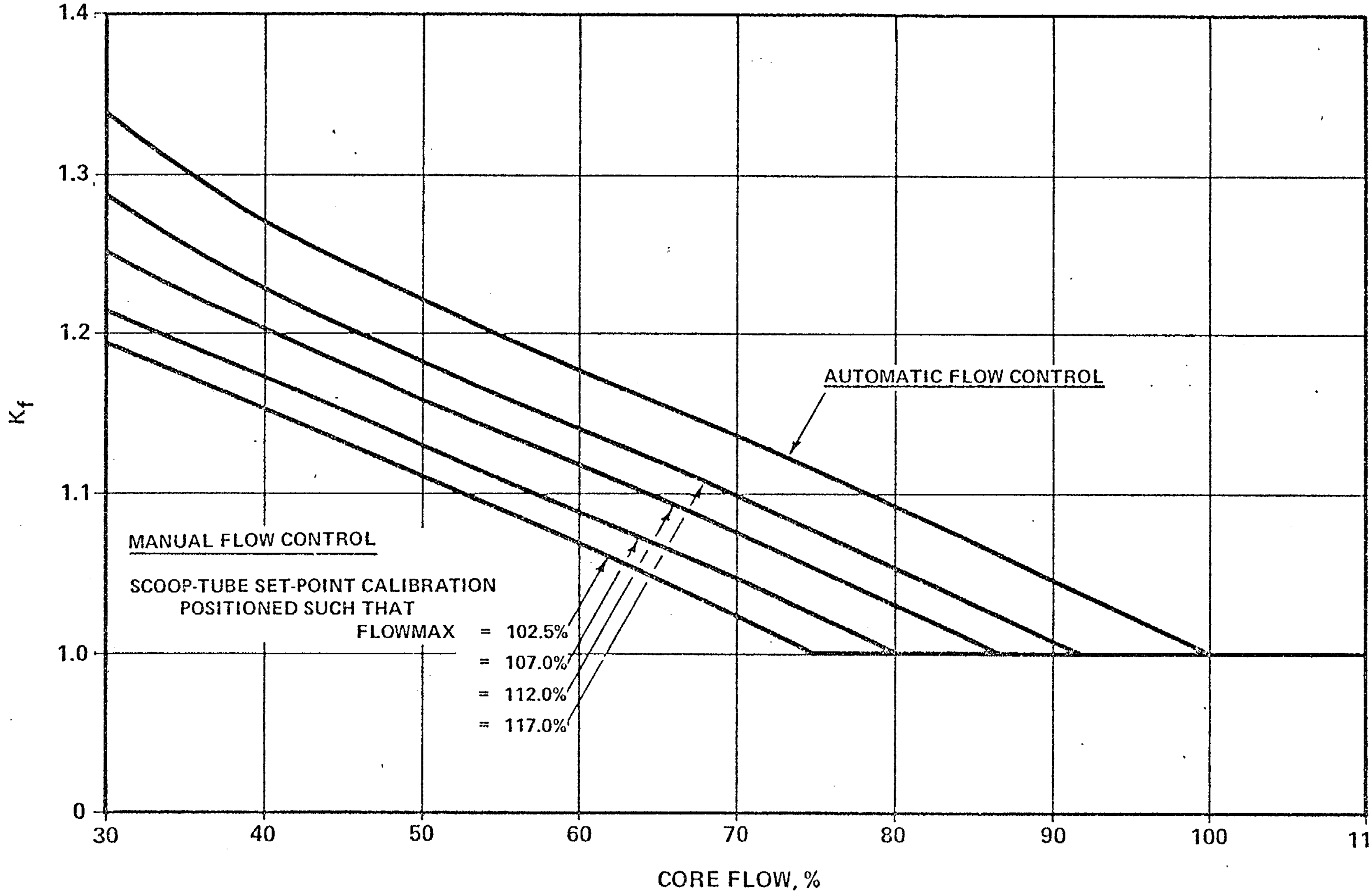

FIGURE 3.11-3
K_F FACTOR

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

NEGATIVE DECLARATION

REGARDING PROPOSED CHANGE TO THE

APPENDIX A TECHNICAL SPECIFICATIONS OF LICENSE NO. DPR-57

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-321

The U. S. Nuclear Regulatory Commission (the Commission) has reviewed the licensee's proposed change to the Appendix A Technical Specifications of Facility Operating License DPR-57. This change would authorize the Georgia Power Company to operate the Edwin I. Hatch Nuclear Plant Unit 1 with certain revisions to the present limiting conditions for operation specified in Appendix A of the referenced license. These revisions result from the implementation of the Acceptance Criteria for the Emergency Core Cooling System for Light Water Nuclear Power Reactors (ECCS) as specified in Section 50.46 of Part 50 CFR. No revisions to the Environmental Technical Specifications, (Appendix B) were required as a result of this proposed change.

The Commission's Division of Reactor Licensing has prepared an environmental impact appraisal for the proposed change to the Appendix A Technical Specifications for Facility Operating License DPR-57.

On the basis of the environmental impact appraisal presented in this document, we have concluded that an environmental impact statement for this particular action is not warranted because, pursuant to the Commission's regulations in 10 CFR 51 and the Council of

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

Environmental Quality's Guidelines, 40 CFR 1500.6, the Commission has determined that this proposed change to the Appendix A Technical Specifications is not a major federal action significantly affecting the quality of the human environment. The environmental impact appraisal is available for public inspection at the Commission's Public Document Room 1717 H Street, N.W., Washington, D. C. 20555, and at the Appling County Public Library, Parker Street, Baxley, Georgia 31513.

Dated at Rockville, Maryland, this 15th day of December, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Chief
Environmental Projects Branch No. 1
Division of Reactor Licensing

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE DIVISION OF REACTOR LICENSING

SUPPORTING: AMENDMENT NO. 27 TO LICENSE NO. DPR-57

CHANGE NO. 26 TO THE TECHNICAL SPECIFICATIONS (APPENDIX A)

GEORGIA POWER COMPANY

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

I. Description of Proposed Action

By letters dated July 9, 1975, August 6, 1975, and December 2, 1975, the Georgia Power Company (the licensee) provided information and supportive analysis relative to a proposed change in Appendix A Technical Specifications of Facility License No. DPR-57. Supplementary information was also provided by the licensee in their submittals of September 24, 1975, October 14, 1975, and December 8, 1975. The proposed change concerns revisions to the limiting conditions for operation to the Edwin I. Hatch Nuclear Plant Unit 1 as a result of the implementation of the Acceptance Criteria for the Emergency Core Cooling System (ECCS).

The Georgia Power Company is currently licensed to operate E. I. Hatch Unit 1 at power levels up to 2436 megawatts thermal (MWt). The staff has estimated that the proposed action will result in an approximate 10 to 15 percent reduction in full power operation at this unit during the remaining period of the present fuel cycle (approximately 15 months).

II. Environmental Impacts of Proposed Action

The NRC has evaluated the potential environmental impacts associated with this proposed license amendment as required by the NEPA and Section 51.7 of Part 51 CFR. The staff has determined that the implementation of the ECCS Acceptance Criteria at E. I. Hatch Unit 1 will not result in a signifcant long term decrease in power level. As such, there will be no resultant changes to cooling water requirements, thermal effluents or chemical and radiological effluents that would significantly impact the environment during normal operation or post-accident conditions. The restriction on heat generation rate will require monitoring and control of fuel utilization; however, the staff anticipates that no reduction in total fuel burnup will result from the revised ECCS Acceptance Criteria and evaluation methods.

No environmental impacts, either radiological or nonradiological are expected other than those presented in the Final Environmental Statement (FES) issued October 1972 for the operation of E. I. Hatch Unit 1. The staff's evaluation of radioactive effluent releases is based upon the total quantity of nuclear fuel within the reactor. The proposed action would not affect the total quantity of fuel used at this facility, and thus no increases in radiation doses to the environment are expected. Futhermore, the staff has concluded that the issuance of this change to the Appendix A Technical Specifications would not significantly affect the cost-benefit balance presented in the FES and would not require changes to the Environmental Technical Specifications.

III. Conclusions and Basis for Negative Declaration

On the basis of the NRC evaluation and information supplied by the licensee, it is concluded that the implementation of the ECCS Acceptance Criteria for the Edwin I. Hatch Nuclear Plant Unit 1 will produce no discernible environmental impacts other than those previously addressed in the FES of October 1972. Having reached this conclusion, the Commission has determined that an environmental impact statement need not be prepared for the proposed license amendment and that a Negative Declaration shall be issued to this effect.

DATED: December 15, 1975

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. DPR-57

(CHANGE NO. 26 TO THE TECHNICAL SPECIFICATIONS)

GEORGIA POWER COMPANY AND OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

EDWIN I. HATCH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-321

1.0 Introduction

Georgia Power Company (GPC) has proposed to operate Edwin I. Hatch Nuclear Plant Unit 1 for the remainder of the current fuel cycle under the following conditions:

(1) with plugged bypass flow holes in the lower core support plate as requested in their submittal dated December 2, 1975, and supplements dated December 8 and December 10, 1975;

(2) using operating limits based on the General Electric Thermal Analysis Basis (GETAB) as requested in their submittal dated July 9, 1975, and supplements dated August 29, September 24, October 14, October 21, December 2, and December 8, 1975; and

(3) using modified operating limits based on an acceptable emergency core cooling system evaluation model that conforms with Section 50.46 of 10 CFR Part 50 as requested in their submittal dated July 9, 1975, and supplements dated August 6, October 14, December 2, and December 8, 1975.

2.0 OPERATION WITH PLUGGED BYPASS FLOW HOLES

2.1 Introduction

On November 20, 1975, the Commission issued Amendment No. 23 to the Facility Operating License for Edwin I. Hatch Nuclear Plant Unit 1, thereby authorizing installation of plugs in the bypass flow holes of the lower core support plate. As discussed in our Safety Evaluation supporting this amendment, installation of plugs in the lower core support plate bypass flow holes is designed to reduce the flow-induced core instrument tube - fuel bundle channel box interaction which has resulted in unacceptable channel box wear.

By letter dated December 2, 1975, Georgia Power Company submitted a safety analysis(1) in support of continued reactor operation with the bypass flow hole plugs installed.

2.2 Evaluation

2.2.1 Nuclear Design

The primary nuclear effect caused by plugging the bypass flow holes is an increased bypass void fraction and a reduction in the average in-channel void fraction. The in- and out-of-channel void fraction changes give a net increase in the core average void fraction.

At steady state conditions, the increased bypass void fraction results in a small reduction in the maximum local peaking factor within a fuel bundle and an increase in the local bundle power calculational uncertainty. Another consequence of the reduced bypass flow is a small reduction in the infinite multiplication factor of uncontrolled fuel.

The presence of voids in the bypass region affects the relationship between the travelling incore probe (TIP) signal and the local bundle power. The TIP signal is reduced by the presence of voids and could lead to an underprediction of the peak heat flux. The relationship of the power in the four bundles surrounding a TIP instrument tube and the TIP signal as a function of bypass voids was determined by the General Electric Company (GE) by performing three group, two-dimensional diffusion theory calculations. A bypass void correction factor was developed for making appropriate corrections in the local bundle power. This correction factor has been programmed into the process computer.

The uncertainty in the local bundle power caused by bypass voids is taken into account in determining the minimum critical power ratio (MCPR) safety limit. The TIP uncertainty introduced by the bypass voids is zero in the bottom half of the core and increases from 4.00% at the core mid-plane to 5.33% at the core exit.

After the bypass flow holes are plugged, the fuel will be placed in its original core location. The following observations can be made:

(1) the control rod worths are not significantly changed and, consequently, the previous results of the control rod drop analysis remain valid,

(2) the shutdown margin will remain the same as previously analyzed,

(3) the standby liquid control system reactivity insertion rate and magnitude will not be affected.

We conclude that the analysis of the nuclear performance of the plant with plugged bypass holes is acceptable.

2.2.2 Mechanical Design

2.2.2.1 Bypass Flow Hole Plugs

The only mechanical design change in the reactor is the use of plugs to fill the bypass flow holes. The plug consists of two stainless steel parts (body and shaft) which are connected by an Inconel spring. The shoulder of the body rests on the top of the core plate along the rim of a one-inch bypass hole and is pressed down by the spring. An equal and opposite force is applied on the shaft. A stainless steel latch is connected to the bottom of the shaft by means of a pin. This latch is free to rotate about the pin and latches the shaft to the core plate. The spring exerts a minimum of 35 pounds on the body and latch and a maximum of 46 pounds (with the worst tolerance combination).

Removal of a plug can be accomplished by applying about 500 pounds of force and deforming the latch plastically. More than 10 plugs were removed in tests performed at the GE test facility with consistent latch deformations without damaging other parts.

Plugs identical to those to be used in the Hatch-1 reactor have been installed in the Vermont Yankee, Duane Arnold, Pilgrim, and Cooper reactors. The plugs installed in Vermont Yankee were removed during a refueling operation after 10 months of successful service. No abnormalities or loose pieces were reported. Vermont Yankee has since reinstalled the plugs.

Pressure differentials across the core plate during normal steady state operation and following a steam line break accident are expected to be on the order of 23 to 32 psi. These loads together with the spring preload will produce yielding of the latch in bending but will be significantly below about 500 pounds of force necessary for removing the plug. The 1973 GE full scale flow mockup test shows that, with up to 40 psi differential pressure, there is negligible leakage flow through the plugged holes. No plug vibration was observed during the test and no apparent deformation on the latch was evident after the test. No fatigue and plastic strain ratcheting is expected since the plant power cycle during the anticipated service period will be minimal.

Stainless steel and Inconel are compatible with other reactor internals and are not expected to introduce any unusual oxidation and stress corrosion problems. The flux level at the core plate elevation is estimated to be quite low and an insignificant reduction in ductility due to irradiation is anticipated. GE has performed creep tests with both Inconel springs and stainless steel latches and found that stress relaxation or creep deformation were insignificant. The tests were performed at 550°F.

Georgia Power Company presented to the NRC staff a summary of channel inspections on BWR-2s and BWR-3s. These older plants have instrument tubes similar to Hatch-1, but no bypass flow holes in the core support plate. The bypass flow for these plants enters through clearances in the assembly end fittings, which is similar to the proposed Hatch-1 configuration with plugged bypass holes. One hundred sixty-four channels (adjacent to instrument tubes and source tubes) were inspected during normal fuel outages in 7 plants. No significant channel wear was observed at the corners adjacent to the instrument tubes.

Based on a review of the design, the test rig, the installation methods and primarily the previously successful operating experience at Vermont Yankee and Pilgrim, we conclude that the plugs will not fail so as to result in loose parts in the core or result in unplugging of the bypass flow holes. Also, we conclude that the installed plugs will substantially reduce the instrument tube vibration, due to flow through the bypass holes, sufficient to preclude any unacceptable wear for at least the proposed fuel cycle.

2.2.2.2 Inspection Program

During the November-December, 1975 outage, Georgia Power Company performed an inspection of all the Hatch Unit 1 fuel bundle channel boxes from locations adjacent to in-core instrument tubes.

This inspection was conducted using the General Electric design criteria for maximum acceptable channel box wastage of 0.010 inches for the lower 80 inches of the channel and 0.020 inches for the remaining length. The inspection revealed 125 channel boxes with an unacceptable amount of wear. These channel boxes were replaced. Sixty-four channel boxes were identified as exhibiting less wear than the amount established for replacement. These channel boxes were reinstalled in the reactor in locations which are not adjacent to core instrument tubes.

Instrument tubes adjacent to channels that exhibited high wear were also inspected. The examination of the core instrument tubes revealed no wear.

Based on the results of the inspection and replacement program, we have concluded that the condition of the installed channel boxes and the in-core instrument tubes are acceptable.

2.2.3 Transient and Accident Analysis

The postulated transients and accidents were reanalyzed for a core configuration with bypass flow hole plugs installed in the lower core support plate. The thermal hydraulic considerations are discussed in the sections of this report concerning the use of the General Electric Thermal Analysis Basis (GETAB) and the analyses required by 10 CFR Part 50, Section 50.46. The overpressure transient analysis is discussed below.

2.2.3.1 Overpressure Transient Analysis

The licensee has reanalyzed the worst case overpressure transient to demonstrate that an adequate margin exists below the ASME code allowable vessel pressure of 110% of vessel design pressure (1375 psig). The transient analyzed was the closure of all main steam line isolation valves with no direct reactor scram. The assumptions used in the analysis were: operation at 105% rated power, end of cycle scram reactivity insertion rate, void reactivity applicable to the current fuel cycle, reactor scram from high neutron flux, and all safety/relief valves operable. The results of the reanalysis indicate that the peak pressure at the vessel bottom would be 1273 psig, which is 102 psig below the code acceptable pressure. In addition, GPC provided the results of a sensitivity study performed for BWR-4 reactors which indicate that an inoperable safety/relief valve would result in less than a 25 psi increase in the peak vessel pressure. Therefore, in the event that a single safety/relief valve is inoperable, a safety margin of 77 psig would still be available.

We find the overpressure analysis to be acceptable on the basis that the sensitivity study for one failed valve demonstrates that considerable margin exists below the code allowable pressure. However, we require that the details of this sensitivity study be made available to the NRC staff as soon as possible for our review of its generic applicability.

2.2.4 Instrument Tube-Channel Box Interaction Surveillance

Excessive instrument tube-channel interaction previously has been determined from the noise level in the LPRM signals. The plugged bypass flow holes are expected to affect the noise content of the LPRM signals. The noise content in the 1.4 to 3 Hz frequency range caused by vibration of the LPRM instrument tube should be reduced relative to the power dependent noise content. Some increase in the boiling noise, 5 to 50 Hz range, is expected because of boiling in the bypass water region.

Before the plant was shutdown in 1975, extensive LPRM time traces, TIP traces, and power spectral density (PSD) calculations were obtained for a number of combinations of power and flow. These data will provide a basis for evaluating the efficiency of plugging the bypass flow holes. After reactor startup, comparison of similar measurements with pre-shutdown data will be made to confirm that the mechanical vibration of the instrument tubes has been substantially reduced.

Georgia Power Company has committed to conduct a post-plugging surveillance program to monitor instrument tube - channel box interaction. This program will include:

(1) impact monitoring using accelerometers installed at representative locations throughout the core,

(2) channel wear monitoring using LPRM and TIP traces, and

(3) channel box inspection at the first refueling outage following installation of the bypass flow hole plugs.

We conclude that the proposed surveillance program is acceptable.

2.3 Conclusion

Based on the above discussions, we conclude that operation of Hatch-1 with the lower core support plate bypass flow holes plugged is acceptable.

3.0 GENERAL ELECTRIC THERMAL ANALYSIS (GETAB)

3.1 Discussion

By letter dated July 9, 1975, and supplement dated December 2, 1975, Georgia Power Company proposed changes to the Technical Specifications, Appendix A to Facility Operating License No. DPR-57 for Edwin I. Hatch Nuclear Plant Unit 1, which incorporate operating limits based on the General Electric Thermal Analysis Basis (GETAB) described in Reference 2.

The December 2, 1975 submittal included a safety analysis based on operation with the lower core support plate bypass flow holes plugged.

The proposed changes involve the adoption of a new transition boiling correlation termed GEXL which would replace the Hench-Levy critical heat flux correlation as the basis for determining the thermal-hydraulic conditions which would result in a departure from nucleate boiling.

One of the safety requirements for light water cooled nuclear reactors is prevention of damage to the fuel cladding. To prevent damage to the fuel cladding, light water cooled reactors must be designed and operated such that during normal operation and anticipated transients the heat transfer rate from the fuel cladding to the coolant are sufficient to prevent overheating of the fuel cladding. Although transition boiling would not necessarily result in damage to boiling water reactor (BWR) fuel rods, historically it has been used as a fuel damage limit because of the large reduction in heat transfer rate when film boiling occurs. A critical power ratio (CPR) is defined which is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest. The minimum critical power ratio (MCPR) is the critical power ratio corresponding to the most limiting fuel assembly in the core. The fuel assembly power at which boiling transition would be predicted to occur, using the GEXL correlation, is termed the critical power. The GEXL transition boiling correlation is more recent than the previously used Hench-Levy critical heat flux correlation and is based on an extensive data base. The methods for applying the GEXL correlation to determine thermal limits has been termed the General Electric Thermal Analysis Basis (GETAB). We have accepted the GEXL correlation and the GETAB methods in a previous report [3] as a basis for establishing the safety limit and limiting conditions for operation related to prevention of fuel damage for General Electric BWR 8 x 8 and 7 x 7 fuel. To apply GETAB to the Technical Specifications involves 1) establishing the fuel damage safety limit, 2) establishing limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and 3) establishing limiting conditions for operation such that the initial conditions assumed in accident analyses are satisfied.

3.2 Evaluation

We have evaluated, and report herein, the thermal-hydraulic margins developed for Hatch Unit 1 which are based on the General Electric Company NEDO-10958 report[2] and the additional plant specific information submitted by the licensee.

3.2.1 Fuel Cladding Integrity Safety Limit

The proposed fuel cladding integrity safety limit is specified by a MCPR of 1.06. This value is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are expected to avoid boiling transition during abnormal operational transients provided that MCPR $\geq$ 1.06. The uncertainties in the core and system operating parameters and in the GEXL correlation, Table 5-1 of the licensee's submittal,[1] when combined with the relative bundle power distribution in the core, form the basis for the GETAB statistical determination of the safety limit MCPR.

These uncertainties are the same as or more conservative than those reported in NEDO-10958[2] and NEDO-20340[4] with one exception. The exception is the uncertainty of the bypass void effect on TIP which accounts for the additional uncertainty due to the bypass void content resulting from plugging the core support plate bypass holes. The reactor core selected for the GETAB statistical analyses is a typical core of the 251/764 design (251" diameter vessel/764 fuel assemblies). This typical core is of the same reactor class as the Hatch Unit 1 core (218/560) but it is larger. The bundle power distribution used for the GETAB application has more high power bundles than the distribution expected during operation of the Hatch 1 reactor. This results in a conservative value of the MCPR safety limit which assures that 99.9% of the rods do not experience boiling transition. Consequently, the GETAB analysis results which provide a fuel cladding integrity safety limit of 1.06 can be conservatively applied to the Hatch 1 core with the bypass flow holes plugged.

We conclude that the proposed fuel integrity safety limit, a MCPR of 1.06, is acceptable for the current Hatch Unit 1 fuel cycle.

3.2.2 Operating Limit MCPR

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not violated during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio (Δ MCPR). Addition of the largest Δ MCPR to the safety limit MCPR gives the minimum operating limit MCPR to avoid violation of the safety limit, should the most limiting transient occur.

The transient analyses were evaluated with the end-of-cycle scram reactivity insertion rates that include a design conservatism factor of 0.80. The initial conditions used for the operational transient analyses are acceptable. The initial MCPR assumed in the transient analyses was equal to or greater than the established operating limit MCPR.

Conservatism was applied in the determination of the required operating limit MCPR because the axial and local peaking were assumed to take place at the beginning of the fuel cycle. That is an R-factor* of 1.081 for the 7 x 7 fuel in the Hatch 1 core and an axial peaking factor of 1.40 at the core midplane were assumed. These assumptions constitute the worst consistent set of parameters that are supported by a General Electric study[2] previously approved by the NRC

*An R-factor is a parameter which characterizes the local peaking pattern with respect to the most limiting rod.

staff. This study demonstrates that the required operating MCPR varies with the axial and local power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. During the fuel cycle the local peaking and therefore the R-factor is reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced end-of-cycle R-factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane. Thus, the effects of the change in the R-factor and the axial peaking factor compensate each other.

3.2.2.1 Core-Wide Transients

The licensee has submitted the results of analyses of those core-wide transients which produce a significant decrease in ΔMCPR.

The types of transients evaluated were losses of flow, pressure and power increases, and coolant temperature decreases. The most limiting transients in these categories were two-pump trip, turbine trip without bypass, and loss of a feedwater heater. Of these three the most limiting transient was the turbine trip without bypass transient which results in a ΔMCPR of 0.26. The calculated ΔMCPR for the second most severe transient, loss of a feedwater heater, is 0.16. Consequently, based upon the analyses of core-wide transients, the minimum required operating limit MCPR is 1.32.

3.2.2.2 Local Transient

The rod withdrawal error transient is the worst local power increase transient and is discussed in reference 1 in terms of worst case conditions. The licensee has submitted an analysis of this transient which shows that the local power range monitor subsystem (LPRM's) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will stop rod withdrawal while the critical power ratio is still greater than the 1.06 MCPR safety limit, and the cladding is under the one percent plastic strain limit. We conclude that the consequences of this localized transient are acceptable.

3.2.2.3 Summary

The minimum required operating limit MCPR is based on the most limiting of the core-wide and local transients, the turbine trip without bypass transient. The resulting MCPR limit is 1.32.

We conclude that sufficient operating restrictions will be imposed on Hatch Unit 1 to assure that the safety limit MCPR will not be violated in the event of an anticipated abnormal transient initiated at or above the minimum required operating limit discussed above.

3.2.3 Operating MCPR Limits for Less Than Rated Power and Flow

The limiting transient at less than rated power and flow condition is the recirculation pump speed control failure. The Technical Specifications would require the licensee to maintain MCPR greater than 1.32 times the K_f factor for core flows less than rated. The K_f factor curves were generically derived and assure that the most limiting transient, a speed control increase, occurring at less than rated flow will not exceed the safety limit MCPR of 1.06. We conclude that the limiting conditions for operation, MCPR, at less than rated power and flow are acceptable.

3.3 Conclusion

Based on the above, we conclude that the analyses and operating limits based on the use of the General Electric Thermal Analysis Basis are acceptable. The associated proposed changes to the Technical Specifications which we also conclude to be acceptable are itemized below.

Section 1.0 Definitions

The subsection which defines peaking factor in terms of fuel rod surface heat fluxes will be replaced by a new subsection which defines a total peaking factor in terms of power profile. Subsections will be added to define Critical Power Ratio (CPR) and Minimum Critical Power Ratio (MCPR). The subsection which defines Minimum Critical Heat Flux Ratio will be deleted. These changes are needed to assure consistency with the revised format of the GETAB analysis.

Section 1.1 Fuel Cladding Integrity Safety Limits

Subsection 1.1.A for operations with reactor pressure greater than 800 psig or core flow greater than or equal to 10% of rated would be revised to state a MCPR safety limit. Subsection 1.1.B would be revised to limit core thermal power to 25% or less of rated thermal power when reactor pressure is less than or equal to 800 psig or core flow is less than 10% of rated. Figure 1.1-1 will also be modified to reflect the abovementioned revisions. These changes are consistent with the GETAB analyses discussed earlier in this safety evaluation.

Section 3.3.C Control Rod Drive System

Subsection 3.3.C.2 which describes control rod drive scram insertion times would be revised to assure that reactor operation conforms to the scram reactivity insertion rate curve which was used in the GETAB analysis.

Section 3.3.F Operation with a Limiting Control Rod Pattern

The existing specification 3.3.F will be revised from a MCHFR limitation to a MCPR limitation so that the specification will be consistent with the GETAB analysis.

Section 3.11.C Minimum Critical Power Ratio (MCPR)

Specification 3.11.C will be changed to replace the existing MCHFR limit on the fuel with an operating MCPR limit of 1.32. This MCPR limit is consistent with the GETAB analyses discussed earlier in this report.

Other Changes

The bases will also be changed to discuss the justification for the revised specification itemized above.

4.0 ECCS APPENDIX K ANALYSIS

4.1 Discussion

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46 "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that "...the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, 50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results.

On July 9, 1975 the licensee submitted an evaluation of the ECCS performance for the design basis piping break for Hatch Unit No. 1 along with an amendment requesting changes to the Technical Specifications for Hatch Unit No. 1 to implement the results of the evaluation. The licensee incorporated further information relating to the details of the ECCS evaluation by letters dated August 6, and October 14, 1975, to show compliance to the 10 CFR 50.46 criteria and Appendix K to 10 CFR Part 50.

On December 2, 1975 Georgia Power Company submitted a revised ECCS analysis for operation with the lower core support plate bypass flow holes plugged.

The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the NRC staff Safety Evaluation Report of the Hatch Nuclear Power Plant dated December 27, 1974.

The background of our review of the General Electric (GE) ECCS models and their application to Hatch Unit 1 is described in the NRC staff Safety Evaluation Report (SER) for that facility dated December 27, 1974 issued in connection with the related Order for Hatch Unit 1. The bases for acceptance of the principal portions of the GE evaluation model are set forth in our Status Report of October 1974. Together, the December 27, 1974 SER and the October 1974 Status Report with its Supplement describe an acceptable ECCS evaluation model and the basis for the NRC staff's acceptance of the model. The Hatch Unit 1 ECCS Reanalysis, evaluated herein, properly conforms to this acceptable model.

4.2 Evaluation

With respect to functioning of ECCS in the post-accident mode, the reflood and refill computations for the Hatch Unit 1 analysis were based on a modified version of the SAFE computer code, with explicit consideration of the staff recommended limitations. These are described in the December 27, 1974 SER. The Hatch evaluation did not attempt to include any further credit for other potential changes which the December 27, 1974 SER indicated were under consideration by GE at that time.

During the course of our review, we concluded that additional individual break sizes should be analyzed to substantiate the break spectrum curves submitted in connection with the evaluation provided in August 1974. We also requested that other break locations be studied to substantiate that the limiting break location was the recirculation line.

Additional analyses[5] (performed on the lead plant, Quad Cities Unit No. 2, and incorporated by reference) supported the earlier submittal which concluded that the worst break was the complete severence of the recirculation line. These additional calculations provided further details with regard to the limiting location and size of break as well as the worst single failure for the Hatch Unit 1 design. The limiting break continues to be the complete severence of the recirculation suction line assuming a failure of the LPCI injection valve.

We have reviewed the evaluation of ECCS performance submitted by Georgia Power Company for the Hatch Unit 1 plant with plugged bypass holes and conclude that the evaluation has been performed wholly in conformance with the requirements of Section 50.46. Therefore, operation of the reactor would meet the requirements of Section 50.46 provided that operation is limited to the maximum planar linear heat generation rates (MAPLHGR) of figure 3.11-1, sheets 1 and 2 of the GPC submittal dated December 2, 1975 and to a minimum critical power ratio (MCPR) greater than 1.17. The abovementioned MAPLHGR curves appear as limiting conditions for operation in Technical Specification 3.11.A. A minimum operating MCPR limit of 1.32 is assurred by Technical Specification 3.11.C.

Certain operating conditions presently allowed are not in conformance with the analysis performed in accordance with Section 50.46. Certain changes must be made to the proposed technical specifications to conform with the evaluation of ECCS performance. An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, continuous reactor operation under such conditions should not be permitted until the necessary analyses have been performed, evaluated and determined acceptable. The reactor may, however, operate for periods up to 24 hours with one recirculation loop out-of-service. This short time period permits corrective action to be taken and minimizes unnecessary shutdowns which is consistent with other Technical Specifications. During this period of time the reactor will be operated within the restrictions of the thermal analysis and will be protected from fuel damage resulting from anticipated transients.

The LOCA analysis assumed all ADS valves operated for small line breaks with HPCI failure. Since the licensee did not provide a LOCA analysis with one ADS valve out of service for small size line breaks the Technical Specifications will not permit continuous operation with any ADS valve out of service, except as with other ECCS equipment one valve may be out of service for seven days.

A Technical Specification has been added which requires the submission of a Reportable Occurrence report for each event involving operation beyond a specified MAPLHGR, LHGR, or MCPR operating limit. If the corrective action described in the associated limiting conditions for operation is taken, a thirty-day written report will be acceptable.

The abovementioned changes to the Technical Specifications are the mutually acceptable results of discussions between the licensee and the NRC staff.

4.3 Conclusion

We conclude that operation of the reactor in accordance with the Technical Specification changes discussed above will assure that the requirements of 10 CFR Part 50, Section 50.46, are met.

5.0 CONCLUSION

Based on our evaluation of reactor operation with plugged bypass holes, we have concluded that because this change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. Based on our evaluation of operating limits based upon GETAB and on an acceptable ECCS evaluation model, we have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. We have also concluded, based on the considerations discussed in this evaluation that all of the activities discussed herein will be conducted in compliance with the Commission's regulations and that the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 17, 1975

6.0 References

(1) "Edwin I. Hatch Nuclear Plant Unit 1 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged," NEDO-21124, November 1975.

(2) "General Electric BWR Thermal Analysis Basis (GETAB) Data Correlation and Design Application," NEDO-10958 and NEDE-10958.

(3) "Review and Evaluation of GETAB (General Electric Thermal Analysis Basis) for BWRs," Division of Technical Review, Directorate of Licensing, United States Atomic Energy Commission, September 1974.

(4) General Electric "Process Computer Performance Evaluation Accuracy," NEDO-20340, and Amendment 1, NEDO-20340-1, dated June 1974 and December 1974, respectively.

(5) Letter transmitting Appendix K ECCS Break Spectrum Analysis for Quad Cities Unit 2, Commonwealth Edison Company, April 8, 1975.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-321

GEORGIA POWER COMPANY
OGLETHORPE ELECTRIC MEMBERSHIP CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 27 to Facility Operating License No. DPR-57 issued to Georgia Power Company and Oglethorpe Electric Membership Corporation which revised Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit 1, located in Appling County, Georgia. The amendment is effective as of its date of issuance.

The amendment authorizes operation of Hatch Unit 1 (1) with the lower core support bypass flow holes plugged, (2) using operating limits based on the General Electric Thermal Analysis Basis (GETAB), and (3) with modified operating limits based on an acceptable evaluation model that conforms with the requirements of Section 50.46 of 10 CFR Part 50 of the Commission's regulations.

The application for the amendment 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 amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with items (2) and (3) above was published in the FEDERAL REGISTER on August 26, 1975 (40 F.R. 37273). No request for a hearing or petition for leave to intervene has filed following notice of the proposed action on items (2) and (3) above. Prior public notice of item (1) above is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the applications for amendment dated July 9, 1975, December 2, 1975, and December 8, 1975, and supplements thereto dated August 6, 1975, August 29, 1975, September 24, 1975, October 14, 1975, October 21, 1975, and December 10, 1975, (2) Amendment No. 27 to License No. DPR-57, with Change No. 26, (3) the Commission's concurrently issued related Safety Evaluation, and (4) the Commission's Negative Declaration dated December 15, 1975, (which is also being published in the FEDERAL REGISTER) and associated 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 31513.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 17th day of December, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing