DMB-016

December 31, 1984

Docket No. 50-321

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

Dear Mr. Beckham:

The Commission has issued Amendment No. 105 to Facility Operating License No. DPR-57 for the Edwin I. Hatch Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 6, 1984, and supplemented by your submittal dated September 6, 1984.

The revision to the TSs makes changes to reflect the hardware modifications to facility and other items that are a part of the "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program" for Edwin I. Hatch Nuclear Plant, Unit 1.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed. In connection with this action, we prepared an "Environmental Assessment and Finding of No Significant Impact" which was sent to you separately.

Sincerely,

"ORIGINAL SIGNED BY:"

Robert A. Hermann, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 105 to License No. DPR-57
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555
January 9, 1985

DISTRIBUTION:
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DOCKET No. 50-321

MEMORANDUM FOR: Docketing and Service Branch
Office of the Secretary of the Commission

FROM: Office of Nuclear Reactor Regulation

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT 1

One signed original of the *Federal Register* Notice identifie... ...losed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (... ...otice are enclosed for your use.

- [ ] Notice of Receipt of Application for Construction Permit(s) and Operatin... License(s).
- [ ] Notice of Receipt of Partial Application for Construction Permit(s) and Facility Lice... ...ime for Submission of Views on Antitrust Matters.
- [ ] Notice of Consideration of Issuance of Amendment to Facility Operating License.
- [ ] Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- [ ] Notice of Availability of NRC Draft/Final Environmental Statement.
- [ ] Notice of Limited Work Authorization.
- [ ] Notice of Availability of Safety Evaluation Report.
- [ ] Notice of Issuance of Construction Permit(s).
- [x] Notice of Issuance of Facility Ope... License(s) or Amendment(s).
- [ ] Order.
- [ ] Exemption.
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Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As stated

OFFICE	ORB#4:DL						
SURNAME	RIngram:cf						
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OFFICIAL RECORD COPY

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. DPR-57

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

   A. The application for amendment by Georgia Power Company, et al., (the licensee) dated February 6, 1984, as supplemented September 6, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-57 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.105 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Gus C. Lainas, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 31, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 105

FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

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

Remove	Insert
1.1-1	1.1-1
1.1-2	1.1-2
1.1-3	1.1-3
1.1-4	1.1-4
1.1-10	1.1-10
1.1-12	1.1-12
1.1-13	1.1-13
1.1-14	1.1-14
1.1-17	1.1-17
Figure 1.1-1	--
3.1-1	3.1-1
3.1-4	3.1-4
3.1-5	3.1-5
3.1-6	3.1-6
3.1-7	3.1-7
3.1-12	3.1-12
3.1-17	3.1-17
3.2-15	3.2-15
3.2-16	3.2-16
--	3.2-16a
3.2-17	3.2-17
3.2-65	3.2-65
3.3-5	3.3-5
3.3-15	3.3-15
3.3-19	3.3-19
Figure 3.6-5	Figure 3.6-5
3.6-10	3.6-10
3.11-1	3.11-1
3.11-2	3.11-2
3.11-2a	3.11-2a
3.11-3	3.11-3
3.11-4a	3.11-4a
3.11-5	3.11-5
3.11-6	3.11-6
--	Figure 3.11-1 (Sheet 6)
--	Figure 3.11-1 (Sheet 7)
Figure 3.11.3	Figure 3.11.3
Figure 3.11.4	Figure 3.11.4
Figure 3.11.5	Figure 3.11.5
Figure 3.11.6	Figure 3.11.6
--	Figure 3.11.7

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 1.07 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 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% 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 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).

c. APRM Flux Scram Trip Settings (Run Mode)

(1) Flow Referenced Simulated Thermal Power Monitor Scram Trip Setting

When the Mode Switch is in the RUN position the APRM flow referenced simulated thermal power scram trip setting shall be:

1.1.D. Reactor Water Level (Hot or 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 > 378 inches above vessel invert when fuel is seated in the core.

2.1.A.1.c.(1) Flow Referenced Simulated Thermal Power Monitor Trip Setting (Run Mode) (Continued)

$S \leq 0.58W + 62\%$
(Not to exceed 117%)

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)

(2) Fixed APRM High High Flux Scram Trip Setting (Run Mode)

The APRM fixed flux scram trip setting shall not be allowed to exceed 120% of rated thermal power.

2.1.A.1.d APRM Rod Block Trip Setting

This section deleted.

2.1.A.2. Reactor Vessel Water Low Level Scram Trip Setting (Level 3)

Reactor vessel water low level scram trip setting (Level 3) shall be ≥ 8.5 inches (narrow range scale).

3. Turbine Stop Valve Closure Scram

Turbine stop valve closure scram trip setting shall be ≤ 10 percent valve closure from full open. This scram is only effective when turbine steam flow is above that corresponding to 30% of rated core thermal power, as measured by turbine first stage pressure.

2.1.A.4 Turbine Control Valve Fast Closure Scram Trip Setting

Turbine control valve fast closure scram trip shall initiate within 30 milliseconds of start of control valve fast closure. Fast closure is sensed by measuring electro-hydraulic control oil line pressure which decreases rapidly upon generator load rejection and just prior to fast closure of the control valves. This scram is only effective when turbine steam flow is above that corresponding to 30% of rated core thermal power as measured by turbine first stage pressure.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

Scram trip setting from main steam line isolation valve closure shall be ≤ 10 percent valve closure from full open. This scram is effective in the Run Mode.

6. Main Steam Line Isolation Valve Closure on Low Pressure

Main steam line isolation valve closure on low pressure at inlet to turbine valves shall occur at ≥ 825 psig, while in the Run Mode.

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

Main steam line isolation valve closure on low condenser vacuum shall occur at ≥ 7 inches Hg vacuum.

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 3-1 of Ref. 8. 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 equal to or greater than the actual operating limit MCPR 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.A.1.a. IRM Flux Scram Trip Setting (Continued)

tism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. 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 1.07. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continues 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 available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer and the Rod Sequence Control System. Worth of individual 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. Generally, 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. This switch occurs when reactor pressure is greater than 825 psig.

c. APRM Flux Scram Trip Settings (Run Mode)

The APRM Flux scram trips in the run mode consist of the flow referenced simulated thermal power monitor scram setpoint and a fixed high-high neutron flux scram setpoint. In the simulated thermal power monitor, the APRM flow referenced neutron flux signal is passed through a filtering network with a time constant which is representative of the fuel dynamics. This provides a flow referenced signal that approximates the average heat flux or thermal power that is developed in the core during transient or steady-state conditions. This prevents spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Examples of events which can result in momentary neutron flux spikes are momentary flow changes in the recirculation system flow, and small pressure disturbances during turbine stop valve and turbine control valve testing. These flux spikes represent no hazard to the fuel since they are only of a few seconds duration and less than 120% of rated thermal power.

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

The APRM flow referenced simulated thermal power monitor scram trip setting at full recirculation flow is adjustable up to 117% of rated power. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 100°F feedwater heating event, than would result with the 120% fixed high neutron flux scram trip. The lower flow referenced scram setpoint therefore decreases the severity (ΔCPR) of a slow thermal transient and allows lower Operating Limits if such a transient is the limiting abnormal operational transient during a certain exposure interval in the cycle.

The APRM fixed high-high neutron flux scram trip, adjustable up to 120% of rated power, does not incorporate the time constant, but responds directly to instantaneous neutron flux. This scram setpoint scrams the reactor during fast power increase transients if credit is not taken for a direct (position) scram, and also serves to scram the reactor if credit is not taken for the flow referenced scram.

2. Reactor Vessel Water Low Level Scram Trip Setting (Level 3)

The trip setting for low level scram is above the bottom of the separator skirt. This level is > 14 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 designated scram trip setting is at least 22 inches below the bottom of 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 1.07 during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below that corresponding to 30% of rated thermal power, as measured by turbine first stage pressure.

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 form 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 bypassed when turbine steam flow is below that corresponding to 30% of rated thermal power, as measured by turbine first stage pressure.

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 Mode Switch is not in the RUN position.

6. Main Steam Isolation Valve Closure on Low Pressure

The low pressure isolation of the main steam lines at 825 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.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.

8. "Average Power Range Monitor, Rod Block Monitor and Technical Specifications Improvement (ARTS) Program for Edwin I. Hatch Nuclear Plant, Units 1 and 2," NEDC-30474-P, December 1983.

3.1 REACTOR PROTECTION SYSTEM (RPS)

Applicability

The Limiting Conditions for Operation associated with the Reactor Protection System apply to the instrumentation and associated devices which initiate a reactor scram.

Objective

The objective of the Limiting Conditions for Operation is to assure the operability of the Reactor Protection System.

Specifications

A. Sources of a Trip Signal Which Initiate a Reactor Scram

The instrumentation requirements associated with each source of a scram signal shall be as given in Table 3.1-1.

The action to be taken if the number of operable channels is not met for both trip systems is also given in Table 3.1-1.

B. Core Maximum Fraction of Limiting Power Density (CMFLPD)

This section deleted.

4.1 REACTOR PROTECTION SYSTEM (RPS)

Applicability

The Surveillance Requirements associated with the Reactor Protection System apply to the instrumentation and associated devices which initiate a reactor scram.

Objective

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the protection instrumentation to assure operability.

Specifications

A. Test and Calibration Requirements for the RPS

RPS instrumentation systems and associated systems shall be functionally tested and calibrated as indicated in Table 4.1-1.

When it is determined that a channel has failed in the unsafe condition, the other RPS channels that monitor the same variable shall be functionally tested immediately before the trip system containing the failure is tripped. The trip system containing the unsafe failure may be placed in the untripped condition during the period in which surveillance testing is being performed on the other RPS channels.

B. Core Maximum Fraction of Limiting Power Density (CMFLPD)

This section deleted.

Table 3.1-1 (Cont'd)

Scram Number (a)	Source of Scram Trip Signal	Operable Channels Required Per Trip System (b)	Scram Trip Setting	Source of Scram Signal is Required to be Operable Except as Indicated Below
5	High Drywell Pressure	2	$\leq$ 2 psig	Not required to be operable when primary containment integrity is not required. May be bypassed when necessary during purging for containment inerting or deinerting.
6	Reactor Vessel Water Level - Low (Level 3)	2	$\geq$ 8.5 inches	
7	Scram Discharge Volume High High Level			Permissible to bypass (initiates control rod block) in order to reset RPS when the Mode Switch is in the REFUEL or SHUTDOWN position.
	a. Float Switches	2	$\leq$71 gallons	
	b. Thermal Level Sensors	2	$\leq$71 gallons	
8	APRM Flow Referenced Simulated Thermal Power Monitor	2	S $\leq$ 0.58 W + 62% (Not to exceed 117%) Tech Spec 2.1.A.1.c(1)	
	Fixed High High Neutron Flux	2	S $\leq$ 120% Power Tech Spec 2.1.A.1.c(2)	
	Inoperative	2	Not Applicable	An APRM is inoperable if there are less than two LPRM inputs per level or there are less than 11 LPRM inputs to the APRM channel.

Table 3.1-1 (Cont'd)

Scram Number (a)	Source of Scram Trip Signal	Operable Channels Required Per Trip System (b)	Scram Trip Setting	Source of Scram Signal is Required to be Operable Except as Indicated Below
8	APRM Downscale	2	≥ 3/125 of full scale	The APRM downscale trip is active only when the Mode Switch is in RUN. The APRM downscale trip is automatically bypassed when the IRM instrumentation is operable and not tripped.
	15% Flux	2	≤ 15/125 of full scale Tech Spec 2.1.A.1.b	The APRM 15% Scram is automatically bypassed when the Mode Switch is in the RUN position.
9	Main Steam Line Radiation	2	≤ 3 times normal background at rated thermal power.	Not required if all steam lines are isolated.
10	Main Steam Line Isolation Valve Closure	4	≤ 10% valve closure from full open Tech Spec 2.1.A.5	Automatically bypassed when the Mode Switch is not in the RUN position. The design permits closure of any two lines without a scram being initiated.
11	Turbine Control Valve Fast Closure	2	Within 30 milliseconds of the start of control valve fast closure Tech Spec 2.1.A.4.	Automatically bypassed when turbine steam flow is below that corresponding to 30% of rated thermal power as measured by turbine first stage pressure.

Table 3.1-1 (Cont'd)

Scram Number (a)	Source of Scram Trip Signal	Operable Channels Required Per Trip System (b)	Scram Trip Setting	Source of Scram Signal is Required to be Operable Except as Indicated Below
12	Turbine Stop Valve Closure	4	≤10% valve closure from full open Tech Spec 2.1.A.3	Automatically bypassed when turbine steam flow is below that corresponding to 30% of rated thermal power as measured by turbine first stage pressure.

Notes for Table 3.1-1

a. The column entitled "Scram Number" is for convenience so that a one-to-one relationship can be established between items in Table 3.1-1 and items in Table 4.1-1.

b. There shall be two operable or tripped trip systems for each potential scram signal. If the number of operable channels cannot be met for one of the trip systems, that trip system shall be tripped. However, one trip signal channel of a trip system may be inoperable for up to two (2) hours during periods of required surveillance testing without tripping the associated trip system, provided that the other remaining channel(s) monitoring that parameter within that trip system is (are) operable.

For SCRAMS 1 thru 7 and 8 APRM 15% Flux, if the number of operable channels is not met for both trip systems; initiate insertion of all control rods capable of being moved by control rod drive pressure and complete their insertion within four (4) hours.

For SCRAM 8 (APRM High Trips, Inoperative, and Downscale), if the number of operable channels is not met for both trip systems; initiate insertion of all control rods capable of being moved by control rod drive pressure and complete their insertion within four hours or reduce power to the IRM range and go to the START & HOT STANDBY position of the Mode Switch within eight hours.

For SCRAMS 9 and 10, if the number of operable channels is not met for both trip systems; reduce turbine load and close main steam line isolation valves within eight hours or initiate insertion of all control rods capable of being moved by control rod drive pressure and complete their insertion within four hours.

For SCRAMS 11 and 12, if the number of operable channels is not met for both trip systems, reduce reactor power to 25% of rated thermal power or less within eight hours.

Table 4.1-1

Reactor Protection System (RPS) Instrumentation Functional Test, Functional Test Minimum Frequency, and Calibration Minimum Frequency

Scram Number (a)	Source of Scram Trip Signal	Group (b)	Instrument Functional Test Minimum Frequency (c)	Instrument Calibration Minimum Frequency
1	Mode Switch in SHUTDOWN	A	Once/Operating Cycle	Not Applicable
2	Manual Scram	A	Every 3 months	Not Applicable
3	IRM High High Flux	C	Once/Week during refueling and within 24 hours of Startup (e)	Once/Week
	Inoperative	C	Once/week during refueling and within 24 hours of Startup (e)	Once/Week
4	Reactor Vessel Steam Dome Pressure - High	D	Once/Month	Once/operating cycle
5	High Drywell Pressure	A	Once/Month(f)	Every 3 months
6	Reactor Vessel Water Level - Low (Level 3)	D	Once/Month (g)	Once/Operating Cycle
7	Scram Discharge Volume High High Level			
	a. Float Switches	A	Once/Month (f)	(h)
	b. Thermal Level Sensors	B	Once/Month (f)	Once/operating cycle
8	APRM Fixed High-High Flux	B	Once/Week (e)	Twice/Week
	Inoperable	B	Once/Week (e)	Twice/Week
	Downscale	B	Once/Week (e)	Twice/Week
	Flow Referenced Simulated Thermal Power Monitor	B	Once/Week (f)	Once/Operating Cycle
	15% Flux	C	Within 24 Hours of Startup (e)	Once/Week

3.1.A.4. Reactor Vessel Steam Dome Pressure - High (Continued)

setting also protects the core from exceeding thermal hydraulic limits as a result of pressure increases from some events that occur when the reactor is operating at less than rated power and flow.

5. High Drywell Pressure

Pressure switch instrumentation for the drywell is provided to detect a loss of coolant accident and initiate the core standby cooling equipment. A high drywell pressure scram is provided at the same setting (≤2 psig) as the core standby cooling systems initiation to minimize the energy which must be accommodated during a loss of coolant accident. The instrumentation is a backup to the reactor vessel water level instrumentation.

6. Reactor Vessel Water Level - Low (Level 3)

The bases for the Reactor Vessel Water Level-Low Scram Trip Setting (Level 3) are discussed in the bases for Specification 2.1.A.2.

7. Scram Discharge Volume High High Level

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this piping is an instrument volume which is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should the discharge volume fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in a slow scram time or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volume which scram the reactor when the volume of water reaches 71 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not able to perform its function adequately.

8. APRM

Three APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one trip logic and APRM's C and E operate contacts in the other trip logic. APRM's B, D and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration.

a. Flow Referenced Simulated Thermal Power Monitor and Fixed High-High Neutron Flux

The bases for the APRM Flow Referenced Simulated Thermal Power Monitor and Fixed High-High Neutron Flux Scram Trip Settings are discussed in the bases for Specification 2.1.A.1.c.

4.1.A. Test and Calibration Requirements for the RPS (Continued)

Group C devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two categories: They are as follows:

i. Passive type indicating devices that can be compared with like units on a continuous reference.

ii. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a drift of .4% could occur and still provide for adequate margin. For the APRM system, drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven (7) days. Calibration on this frequency assures plant operation at or below thermal limits.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. This is compensated for in the APRM system by calibrating twice a week using heat balance data and by calibrating individual LPRM's every 1000 effective full power hours using TIP traverse data.

Group D devices consist of analog transmitters, master trip units, slave trip units, and other accessories. The general description of the ATTS devices is provided in Reference 3. As evidenced by NEDO-21617-A, the NRC has approved the following surveillance frequencies for ATTS equipment:

1. Once per shift for channel check
2. Once per month for channel functional test
3. Once per operating cycle for channel calibration

B. Maximum Fraction of Limiting Power Density (MFLPD)
This section deleted.

Table 3.2-7

NEUTRON MONITORING INSTRUMENTATION WHICH INITIATES CONTROL ROD BLOCKS

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System	Trip Setting	Remarks
1	SRM	Inoperative	2(b)(c)(d)	Not applicable	Inoperative trip produced by switch not in operate, power supply voltage low, and circuit boards not in circuit.
		Not fully inserted	2(b)(c)(d)	Not fully inserted	This function is bypassed when the count rate is > 100 cps or the IRMs are on range 3 or above.
		Downscale	2(b)(c)(d)	≥ 3 counts/sec	This function is bypassed when the count rate is > 100 cps or the IRMs are on range 3 or above.
		Upscale	2(b)(c)(d)	≤ 10^5 counts/sec.	
2	IRM	Inoperative	3(b)(d)	Not applicable.	Inoperative trip produced by switch not in operate, power supply voltage low, or circuit boards not in circuit.
		Not fully inserted	3(b)(d)	Not fully inserted	Only required in the Refuel and Start & Hot Standby Modes.
		Downscale	3(b)(d)	≥ 5/125 of full scale	Trip bypassed when IRM on Range 1.
		High Flux	3(b)(d)	≤108/125 of full scale	
3	APRM	Inoperative	2(b)(e)	Not applicable	Inoperative trip produced by switch not inoperate, power supply voltage low, or circuit boards not in circuit.

Table 3.2-7 (Continued)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System	Trip Setting	Remarks
3	APRM	Downscale	2(b)(e)	≥ 3/125 of full scale	Not required while performing low power physics test at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
		12% Flux	2(b)(e)	≤ 12/125 of full scale	This function is bypassed when the Mode Switch is placed in the RUN position.
3		Upscale	2(b)(e)	≤ 0.58 W + 50%	W is the loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated power. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.
4	RBM	Inoperative	1(e)(f)	Not applicable	Inoperative trip produced by switch not in operate, circuit boards not in circuit, fails to null, less than required number of LPRM inputs for rod selected.
		Downscale	1(e)(f)	≥94/125 of full scale	

Table 3.2-7 (Continued)

Ref. No. (a)	Instrument	Trip Condition Nomenclature	Required Operable Channels per Trip System	Trip Setting	Remarks
4	RBM	Upscale	1(e)(f)		
		Low Trip Setpoint (LTSP)		≤ 117/125 of full scale	There are three upscale trip levels. Only one is applied over a specified operating core thermal power range. All RBM trips are automatically bypassed below the low power setpoint. The upscale LTSP is applied between the low power and the intermediate power setpoints. The upscale ITSP is applied between the intermediate power setpoint and the high power setpoint. The upscale HTSP is applied above the high power setpoint.
		Intermediate Trip Setpoint (ITSP)		≤ 111.2/125 of full scale	
		High Trip Setpoint (HTSP)		≤ 107.4/125 of full scale	
		Power Range Setpoints	Not applicable		
		Low Power Setpoint (LPSP)		≤ 30% rated core thermal power	Power range setpoints control the enforcement of the appropriate upscale trips over the proper core thermal power ranges. The power signal to the RBM is provided by the APRM.
		Intermediate Power Setpoint (IPSP)		≤ 65% rated core thermal power	
		High Power Setpoint (HPSP)		≤ 85% rated core thermal power	
		Bypass Time Delay (td_2)	Not applicable	≤ 2.0 seconds	RBM bypass time delay is set low enough to assure minimum rod movement while upscale trips are bypassed.
5	Scram Discharge Volume	High Water Level	1(g)	≤ 18 gallons	

Notes for Table 3.2-7

a. The column entitled "Ref. No." is only for convenience so that a one-to-one relationship can be established between items in Table 3.2-7 and items in Table. 4.2-7.

b. For the START & HOT STANDBY position of the Mode Switch, there shall be two operable or tripped systems for each potential trip condition. If the requirements established by the column cannot be met for one of the two trip systems, the condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the requirements established by this column cannot be met for both trip systems, the systems shall be tripped.

c. One of the four SRM inputs may be bypassed.

d. The SRM and IRM blocks need not be operable in the Run Mode. This function is bypassed when the Mode Switch is placed in the RUN position.

e. The APRM and RBM rod blocks need not be Operable in the Start & Hot Standby Mode (Except 12% APRM Rod Block).

f. The RBM is only required when core thermal power is $\geq$ 30% and the limiting condition defined in Section 3.3.F exists.

g. This trip is Operable in Power Operation and Hot Standby Mode, and Refuel Mode when any control rod is withdrawn. Not applicable to control rods removed per Specification 3.10.E.

3.2.G.3.b. Downscale

A downscale indication of ≤ 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 ≥ 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. Upscale (Flow Referenced)

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

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 does not significantly increase the risk of an inadvertent control rod withdrawal. The RBM function is required only when a limiting rod pattern for RWE (Section 3.3.F) exists.

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. Upscale

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

3.3.F. Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

A Limiting Rod Pattern for RWE exists when:

1. Thermal power is below 90% of rated and the MCPR is less than 1.70, or

2. Thermal power is 90% of rated or above and the MCPR is less than 1.40.

During operation with a Limiting Control Rod Pattern for RWE and when core thermal power is ≥ 30%, either:

1. Both RBM channels shall be operable, or

2. If only one RBM channel is operable, control rod withdrawal shall be blocked within 24 hours, or

3. If neither RBM channel is operable, control rod withdrawal shall be blocked.

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 (for Rod Withdrawal Error, RWE)

During operation when a Limiting Control Rod Pattern for RWE exists and only one RBM channel is operable, an instrument functional test of the RBM shall be performed prior to withdrawal of the control rod(s). A Limiting Rod Pattern for RWE is defined by 3.3.F.

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.F Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)

Surveillance Requirements:

A limiting control rod pattern for RWE is a pattern which, due to unrestricted withdrawal of any single control rod, could result in violation of the MCPR Safety Limit. Specification 3.3.F defines a limiting control rod pattern for RWE. During use of such patterns when both RBM channels are not operable, it is judged that testing of the RBM system prior to withdrawal of control rods to assure its operability will assure that improper withdrawal does not occur. Reference NEDC-30474-P (Ref. 17) for more information.

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 characteristics 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-24040.

3.3.J. References (Continued)

12. FSAR Section 7.14.5.3, Rod Worth Minimizer Function

13. FSAR Section 7.7.5, Rod Sequence Control System

14. FSAR Section 3.6.4.1, Control Rods

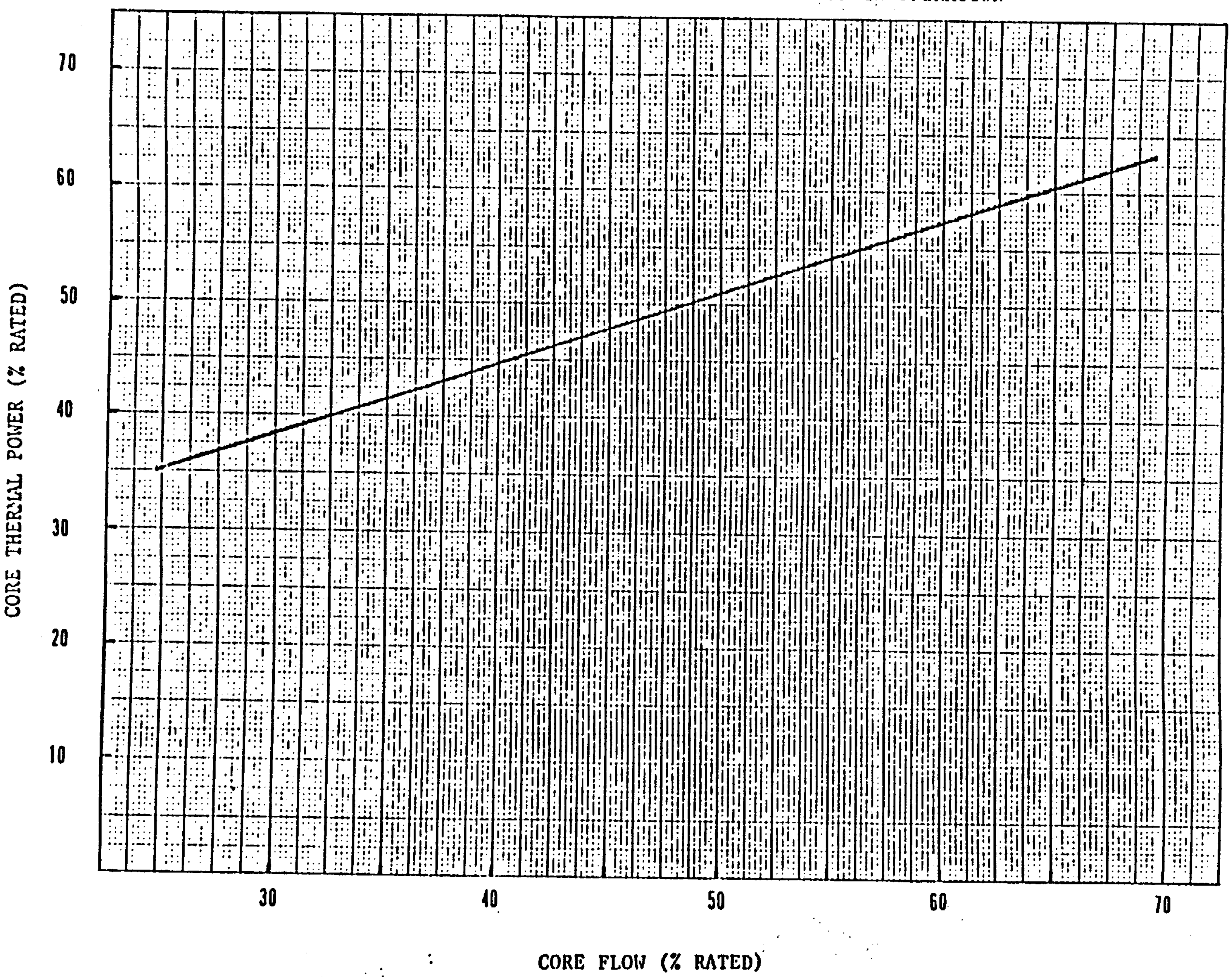
15. FSAR Question 3.6.7, Amendment 24

16. FSAR Appendix P, Rod Sequence Control System (RSCS)

17. "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Edwin I. Hatch Nuclear Plant, Units 1 and 2," NEDC-30474-P, December 1983.

FIGURE 3.6-5

THERMAL POWER LIMITATIONS DURING OPERATION WITH LESS THAN TWO REACTOR COOLANT SYSTEM RECIRCULATION LOOPS IN OPERATION



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. Core thermal power shall not exceed 1% of rated thermal power without forced recirculation.

2. Operation with a single recirculation pump is permitted for 24 hours unless the recirculation pump is sooner made operable. With one recirculation pump not in operation, initiate action within 15 minutes or continue action to reduce reactor power to or below the limit specified in Figure 3.6-5 within 2 hours. If the pump cannot be made operable or the limit of Figure 3.6-5 cannot be met within the required time, the reactor shall be in cold shutdown within 24 hours.

3. 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.

4.6.J. Recirculation Pump Speeds

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

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.

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 NRC.

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 all core locations shall not exceed the appropriate APLHGR limit for those core locations. The APLHGR limit, which is a function of average planar exposure and fuel type, is the appropriate value from Figure 3.11-1, sheets 1 through 5, multiplied by the smaller of the two MAPFAC factors determined from Figure 3.11-1, sheets 6 and 7. 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, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

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 for 7 x 7 fuel or the limiting value of 13.4 kw/ft for 8 x 8/ 8 x 8R fuel. If at any time during

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 ≥ 25% rated thermal power.

B. Linear Heat Generation Rate (LHGR)

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

3.11.B Linear Heat Generation Rate (LHGR) (Continued)

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 LHGR is not returned to within the prescribed limits within two (2) hours, then reduce reactor power to less than 25% of rated thermal power within the next four (4) hours. If the limiting condition for operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

C. Minimum Critical Power Ratio (MCPR)

The minimum critical power ratio (MCPR) shall be equal to or greater than the operating limit MCPR (OLMCPR), which is a function of scram time, core power, and core flow. For 25% ≤ power < 30%, the OLMCPR is given in Figure 3.11.7. For power ≥ 30%, the OLMCPR is the greater of either:

1. The applicable limit determined from Figure 3.11.3, or
2. The applicable limit from either Figures 3.11.4, 3.11.5, or 3.11.6, multiplied by the K_p factor determined from Figure 3.11.7, where:

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

τ_A = 0.90 sec (Specifications 3.3.C.2.a scram time limit to 20% insertion from fully withdrawn)

$$\tau_B = 0.710+1.65\left[\frac{N_1}{\sum_{i=1}^{n} N_i}\right]^{1/2}(0.053) \text{ [Ref.10]}$$

4.11.C.1 Minimum Critical Power Ratio (MCPR)

MCPR shall be determined to be equal to or greater than the applicable limit, daily during reactor power operation at ≥ 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.

4.11.C.2 Minimum Critical Power Ratio Limit

The MCPR limit at rated flow and rated power shall be determined for each fuel type, 8X8R, P8X8R, 7X7 from figures 3.11.4, 3.11.5, and 3.11.6 respectively using:

a. τ=1.0 prior to initial scram time measurements for the cycle, performed in accordance with specifications 4.3.C.2.a

or

b. τ as defined in specification 3.11.C.

The determination of the limit must be completed within 72 hours of the conclusion of each scram time surveillance test required by specification 4.3.C.2.

3.11.C. Minimum Critical Power Ratio (MCPR) (Continued)

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

n = number of surveillance tests performed to date in cycle

N_i = number of active control rods measured in the i^{th} surveillance test

τ_i = Average scram time to 20% insertion from fully withdrawn of all rods measured in the i^{th} surveillance test, and,

N_1 = total number of active rods measured in 4.3.C.2.a.

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, then reduce reactor power to less than 25% of rated thermal power within the next four(4) hours. If the Limiting Condition for Operation is restored prior to expiration of the specified time interval, then further progression to less than 25% of rated thermal power is not required.

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.

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 10 CFR 50, 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 ± 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 conform to 10 CFR 50.46. The limiting value for APLHGR at rated conditions is shown in Figures 3.11.1. sheets 1 thru 5.

A flow dependent correction factor incorporated in to Figure 3.11-1 (sheet 7) is applied to the rated conditions APLHGR to assure that the 2200°F PCT limit is complied with during LOCA initiated from less than rated core flow. In addition, other power and flow dependent corrections given in Figure 3.11-1 (sheets 6 and 7) are applied to the rated conditions APLHGR limits to assure that the fuel thermal-mechanical design criteria are met during abnormal transients initiated from off-rated conditions.

The calculational procedure used to establish the APLHGR shown in Figures 3.11.1, sheets 1 thru 5, is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis as compared to previous analyses performed with Reference 1 are: (1) The analyses assume a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figure 3.11.1; (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limiting as described in Reference 2, are included in the reflooding calculations.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 1 of NEDO-21187[3]. Further discussion of the APLHGR bases is found in NEDC-30474-P[11].

3.11.C. Minimum Critical Power Ratio (MCPR) (Continued)

The purpose of the $MCPR_f$, and the K_p of Figures 3.11.3 and 3.11.7, respectively, is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power, the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

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

The core power dependent MCPR operating limit $MCPR_p$ is the power rated flow MCPR operating limit multiplied by the K_p factor given in Figure 3.11.7.

The K_ps are established to protect the core from transients other than core flow increases, including the localized event such as rod withdrawal error. The K_ps were determined based upon the most limiting transient at the given core power level. (For further information on MCPR operating limits for off-rated conditions, reference NEDC-30474-P.[11])

3.11.C. Minimum Critical Power Ratio (MCPR) (Continued)

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 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. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566-P, November, 1975.

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

3. Edwin I. Hatch Nuclear Plant Unit 1 Emergency Core Cooling System Analysis - Appendix K Requirement With Modified Low Pressure Coolant Injection System, NEDO-21187, Supplement 1, April, 1976.

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

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

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

7. "Edwin I. Hatch Nuclear Plant Unit 1 Channel Inspection and Safety Analysis with Bypass Flow Holes Plugged", NEDO-21124-1, July, 1976.

8. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February, 1973 (NEDO-10802).

9. General Electric Boiling Water Reactor Reload No. 1 Licensing Amendment for the Edwin I. Hatch Nuclear Plant Unit 1 Full Core Drilled Conditions, NEDO-21580, February, 1977.

10. Letter from R. H. Buchholz (G. E.) to P. S. Check (NRC), "Response to NRC request for information on ODYN computer model", September 5, 1980.

11. "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Edwin I. Hatch Nuclear Plant, Units 1 and 2," NEDC-30474-P, December 1983.

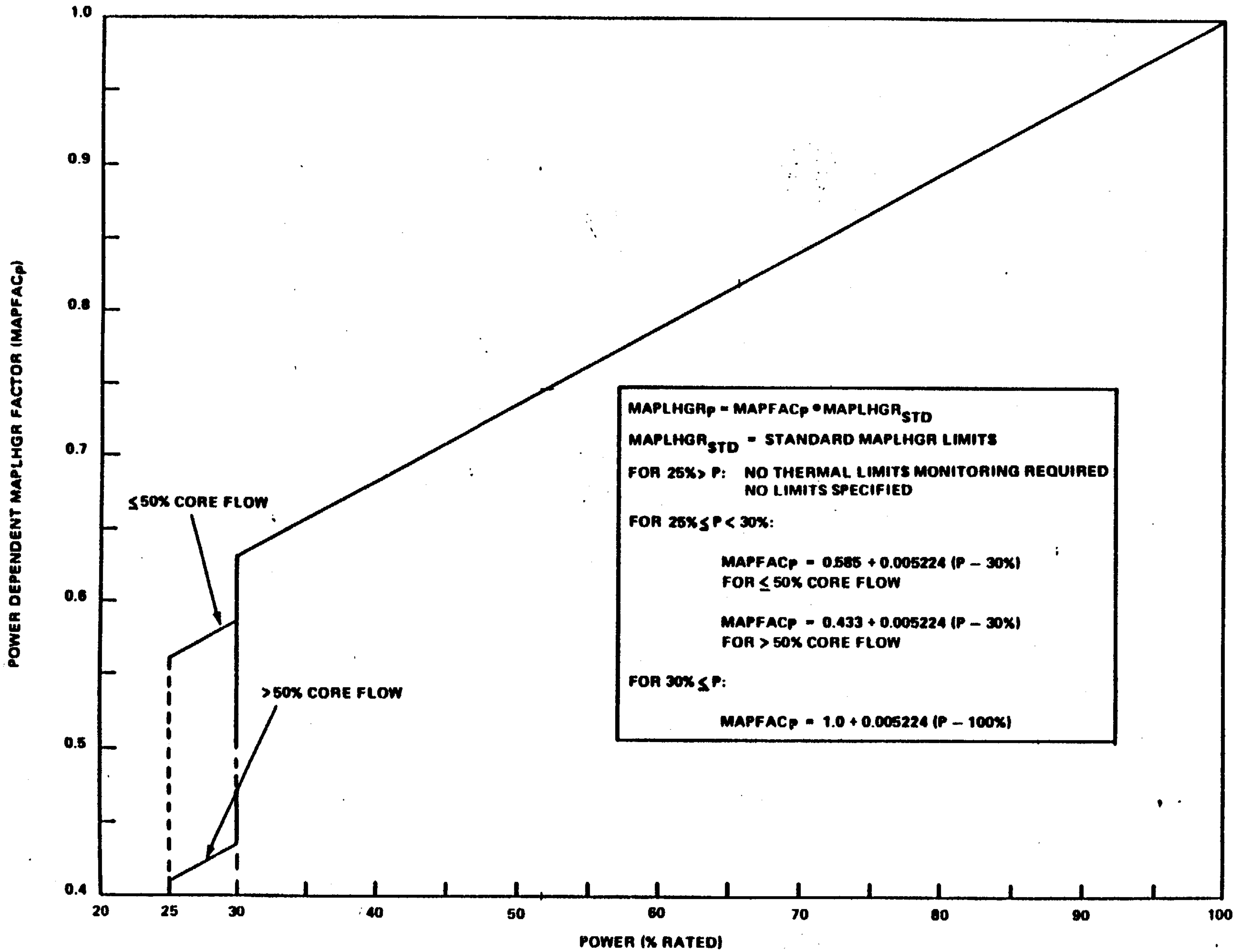
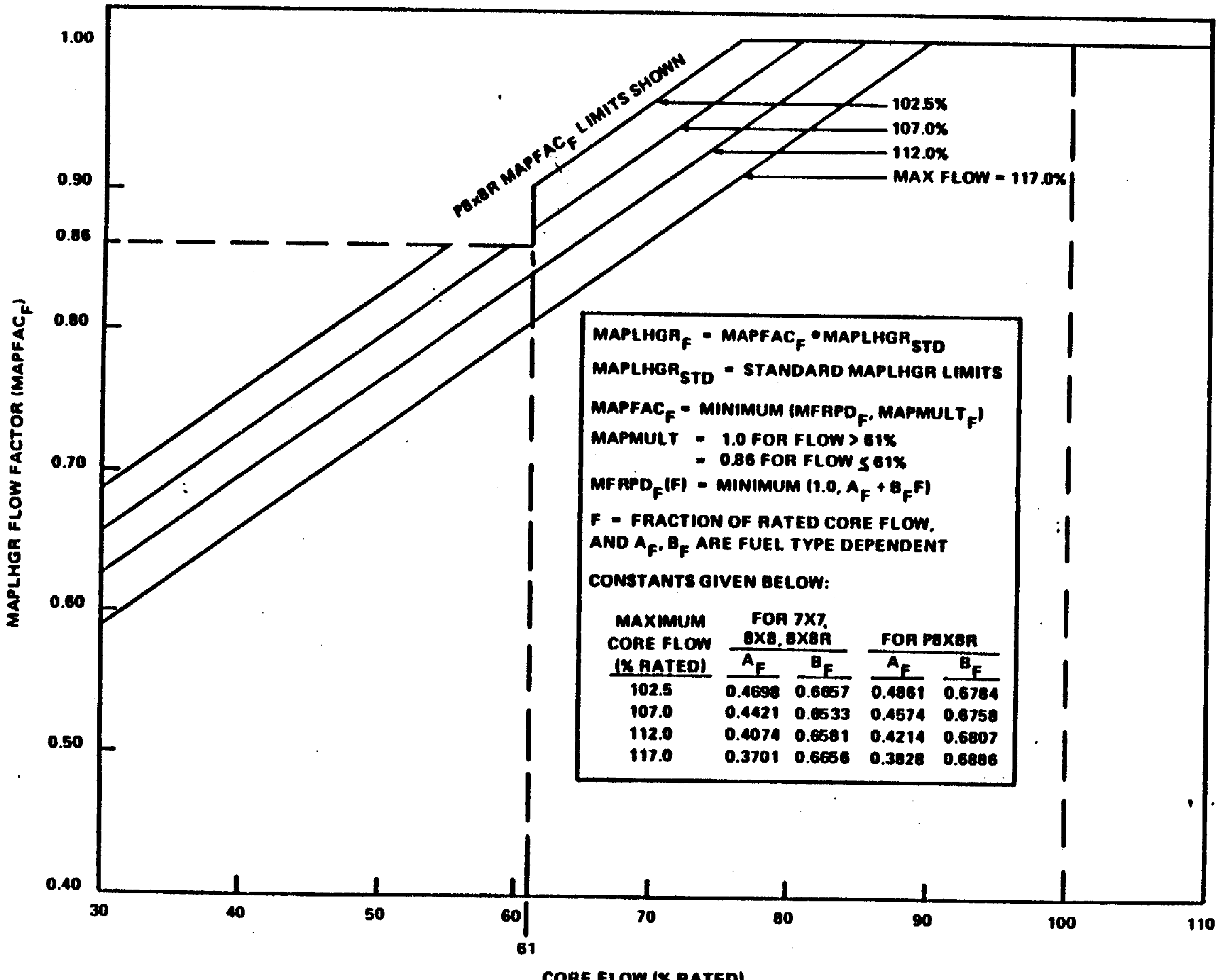


FIGURE 3.11-1 (SHEET 6) $MAPFAC_P$

Amendment No. 105



MAXIMUM CORE FLOW (% RATED)	FOR 7X7, 8X8, 8X8R A_F	FOR 7X7, 8X8, 8X8R B_F	FOR P8X8R A_F	FOR P8X8R B_F
102.5	0.4698	0.6657	0.4861	0.6784
107.0	0.4421	0.6533	0.4574	0.6758
112.0	0.4074	0.6581	0.4214	0.6807
117.0	0.3701	0.6656	0.3828	0.6886

FIGURE 3.11-1 (SHEET 7) MAPFAC$_F$

Amendment No. 105

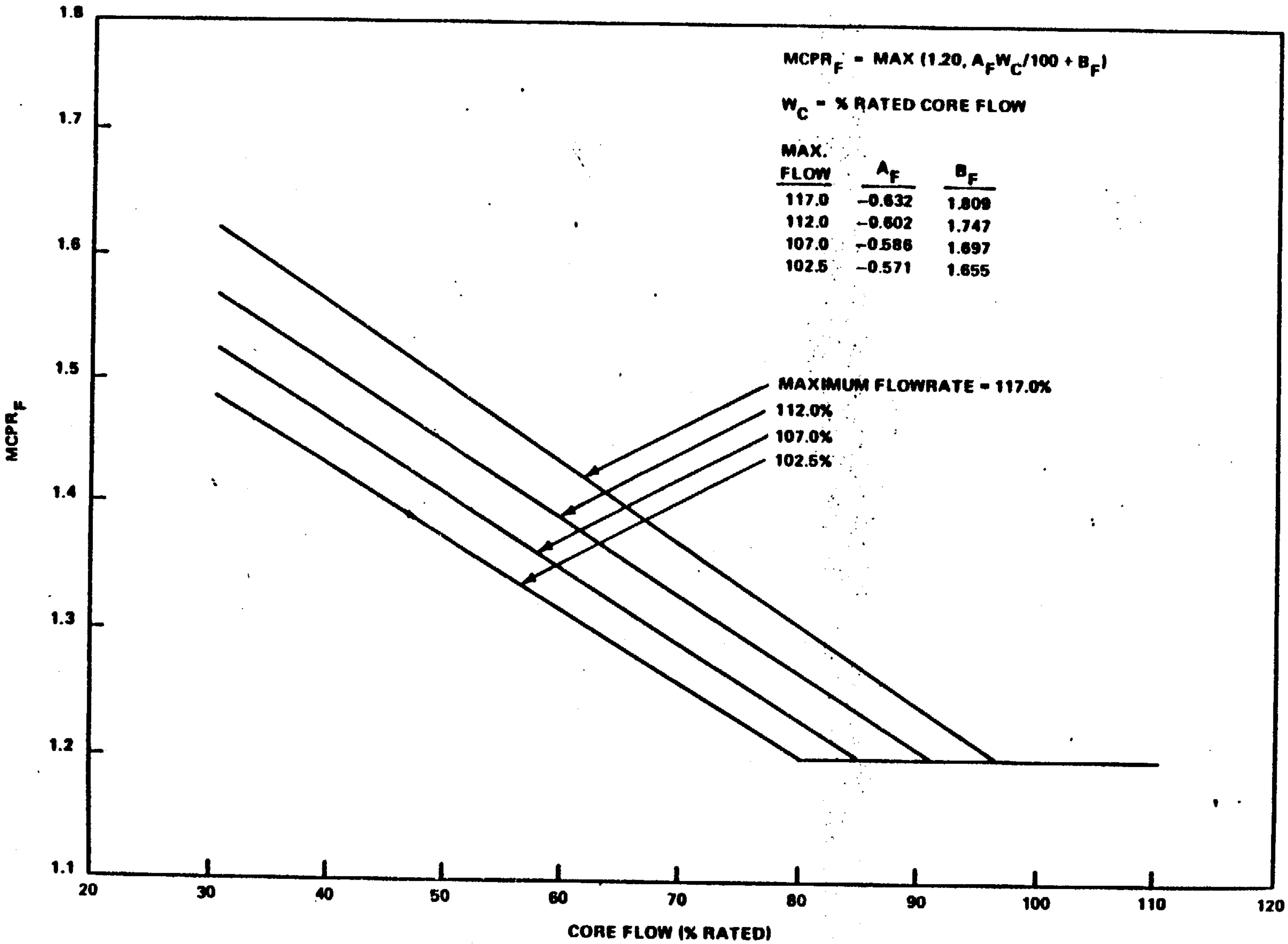


FIGURE 3.11.3 $MCPR_F$

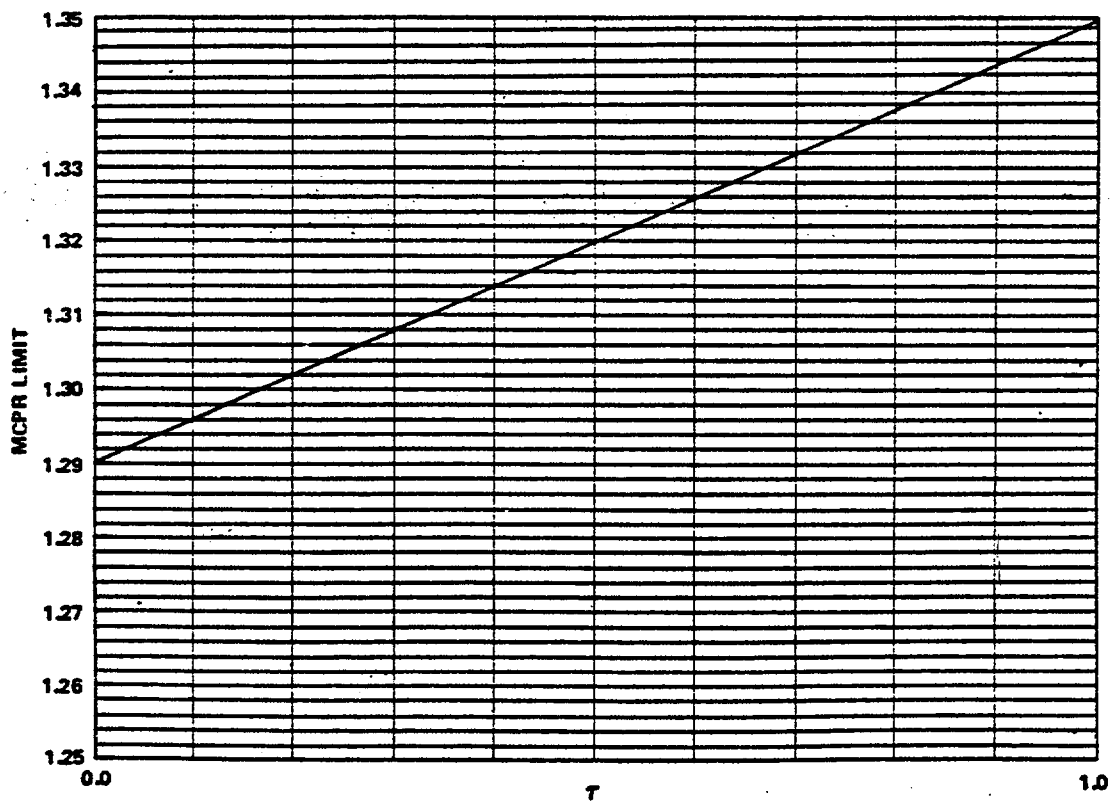


FIGURE 3.11.4
MCPR LIMIT FOR 8X8R FUEL
FOR RATED POWER AND RATED FLOW

Amendment No. ~~86~~, ~~87~~, ~~96~~, 105

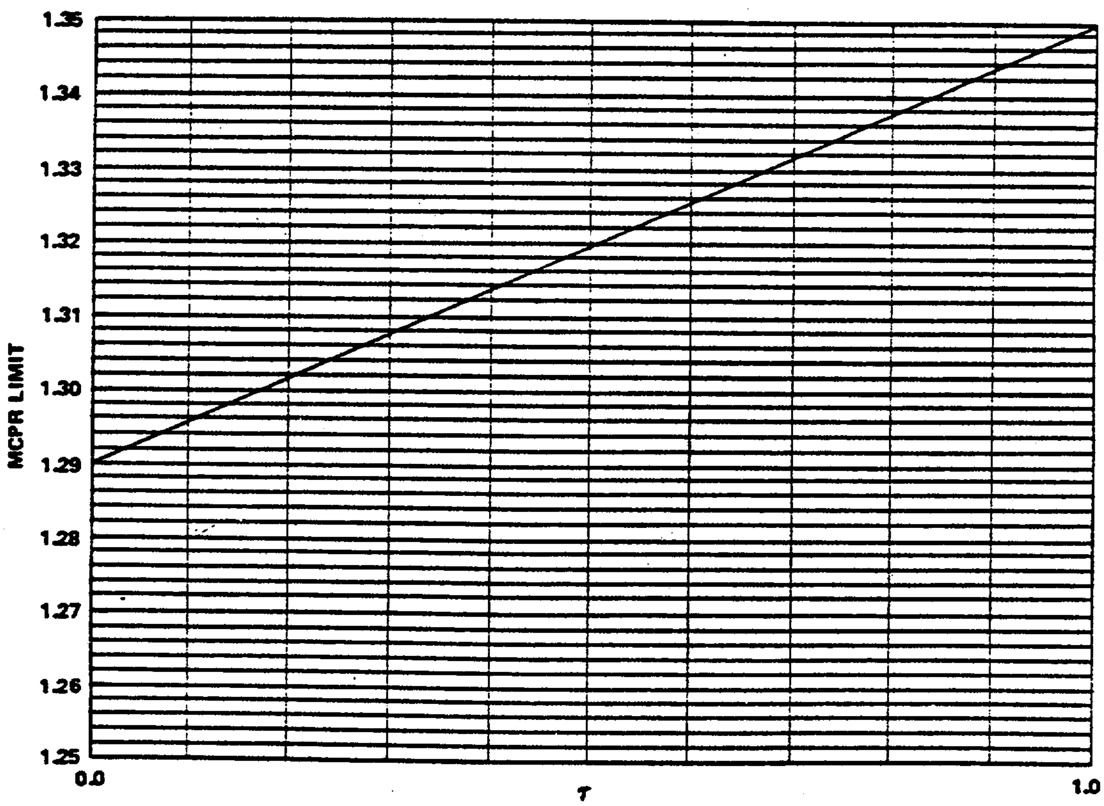


FIGURE 3.11.5
MCPR LIMIT FOR P8X8R FUEL
FOR RATED POWER AND RATED FLOW

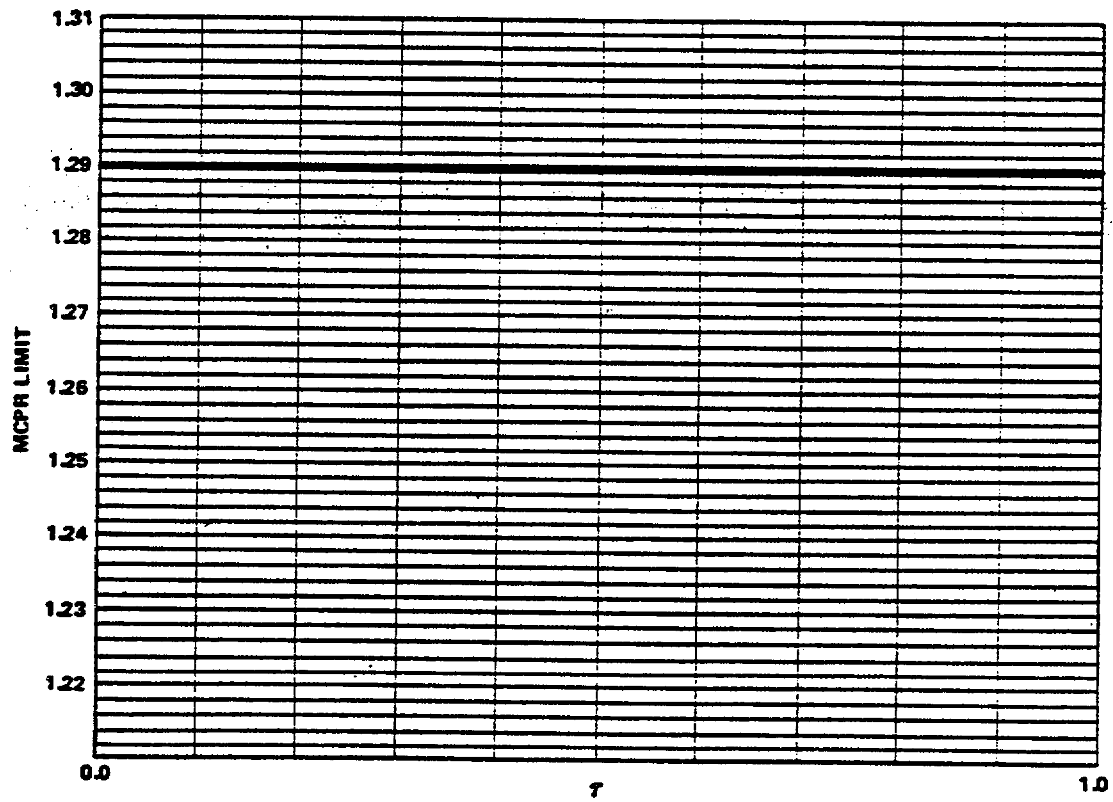


FIGURE 3.11.6
MCPR LIMIT FOR 7X7 FUEL
FOR RATED POWER AND RATED FLOW

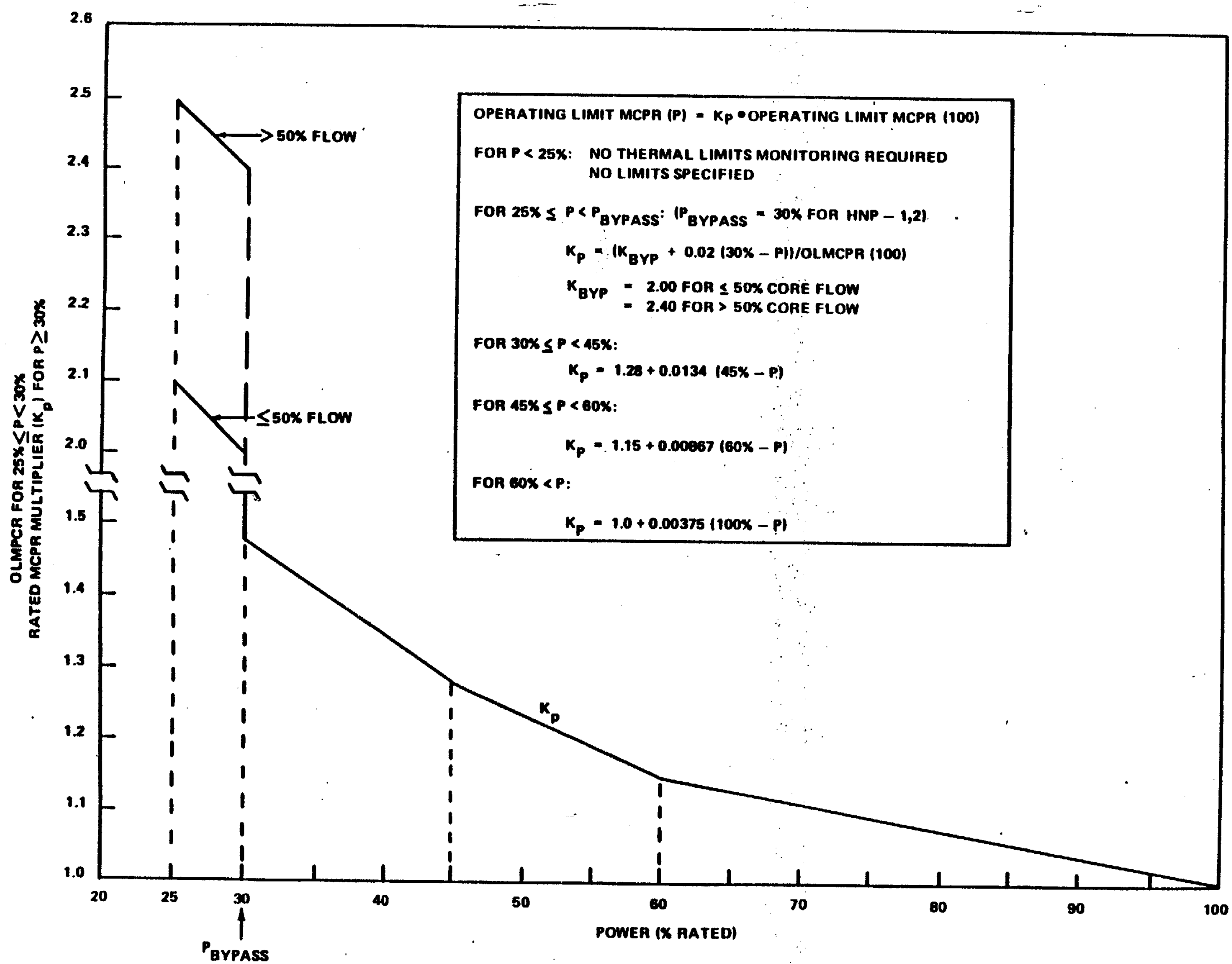


FIGURE 3.11.7 K_p

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. DPR-57

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

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-321

1.0 Introduction

By application dated February 6, 1984, Georgia Power Company (GPC or the licensee) requested an amendment to Appendix A of Facility Operating Licenses Nos. DPR-57 and NPF-5 for Edwin I. Hatch Nuclear Plant, Units 1 and 2, to reflect certain proposed plant modifications related to the Average Power Range Monitor and Rod Block Monitor Instrumentation. We issued the requested changes for Unit 2 in Amendment No. 39 to Facility Operating License No. NPF-5 on July 13, 1984. At that time, the change for Unit 1 was held in abeyance. The licensee, by letter dated September 6, 1984, revised their original request to: 1) provide an additional restriction on plant operation during the Limiting Condition for Operation with one recirculation loop out of service; 2) remove the one-hour time period allowed prior to blockage of control rod withdrawal with the plant in a Limiting Control Rod Pattern and both Rod Block Monitor channels inoperable; and 3) correct a typographical error in the February 6, 1984, submittal.

2.0 Evaluation

The Average Power Range Monitor, Rod Block Monitor, and Technical Specification Improvement (ARTS) Program for Edwin I. Hatch Nuclear Plant, Units 1 and 2, supporting document GE Licensing Report NEDC-30474-P and the associated changes to the Technical Specifications were reviewed and approved in the supporting Safety Evaluation to Amendment 39 to Facility Operating License No. NPF-5 dated July 13, 1984. This evaluation addresses the balance of the changes requested by the licensee in their September 6, 1984, submittal.

2.1 Restricted Plant Operation With One Recirculation Loop Out of Service

The possibility of thermal-hydraulic instability in a BWR has been investigated in the startup of early BWRs. For modern higher-power density reactors, pressure perturbation techniques were developed to measure core stability margins. Based on these tests and analytical models, it has been shown that the high power/low flow corner of the power/flow MAP is the region of least stability margin. This region is encountered during single loop and natural circulation operation of a BWR. The NRC staff has been studying BWR thermal-hydraulic stability characteristics for several years. To assure compliance with General Design Criteria 10 and 12, natural circulation operation has been prohibited and single loop operation has been restricted by Technical Specifications for most plants. GE recently presented the staff with stability test data which demonstrated the occurrence of limit cycle neutron flux oscillations at natural circulation and several percent above the rated rod line.

The oscillations were observable on the APRM's and were suppressed with control rod insertion. It was predicted that limit cycle oscillations would occur at the operating condition tested; however, the characteristics of the observed oscillations were different than those previously observed during other stability tests. Namely, the test data show that some LPRM indications oscillated out of phase with the APRM signal and at an amplitude as great as six times the core average. GE has prepared and released a Service Information Letter, SIL-380, to the licensees with measures to avoid and control abnormal neutron flux oscillations. The major action recommended by SIL-380 to be taken by utilities to avoid the regions of least stability following a recirculation pump(s) trip event is to reduce power by inserting control rods to or below the 80% rod line using the plant's prescribed control rod shutdown insertion sequence.

In response to NRC staff concerns and to make the Technical Specifications consistent with the operating recommendations of SIL-380, GPC, in a letter dated September 6, 1984, has proposed changes to the Edwin I. Hatch Unit 1 Technical Specifications.

The principal addition made to the Technical Specifications is the following:

> When operating with one recirculation loop, the plant will initiate within 15 minutes an orderly reduction in thermal power to less than a specified limit within 2 hours. This limit corresponds to a load line leading to 80% reactor power at rated core flow.

We have reviewed these proposed changes and have found that they result in a considerably more stable operating mode since the plant will be operating at a lower power/flow ratio which has been shown by testing and analysis to result in increased thermal-hydraulic stability. We find that these changes are prudent and acceptably resolve our thermal-hydraulic stability concerns for Hatch Unit 1 since long term single loop operation is not permitted and natural circulation operation is prohibited. Should such operation be requested in the future, we would reevaluate this Technical Specification to determine if additional changes are necessary.

2.2 Clarification of the Limiting Condition for Operation with a Limiting Control Rod Pattern

As a part of the ARTS submittal of February 6, 1984, the licensee proposed changes to Section 3.3.F of the Technical Specifications, "Operation with a Limiting Control Rod Pattern (for Rod Withdrawal Error, RWE)." The NRC staff and licensee discussed the proposed change and mutually agreed that the change to the paragraph was acceptable with the exception of "within 1 hour" in sentence 3 of the paragraph on operation with a Limiting Control Rod Pattern for RWE when core thermal power is $\geq$30%. In order to clarify this issue, the licensee revised the sentence by deleting the "within 1 hour" in their submittal of September 6, 1984. We have reviewed this change and find that it is acceptable since the change makes it clear that control rod withdrawal is not allowed without at least one operable RBM channel when a limiting control rod pattern exists.

2.3 Administrative Error - Technical Specification Table 3.2-7

The licensee, in their submittal of September 6, 1984, identified that "$\leq$" was omitted for the trip setting for the Low Trip Setpoint (LTSP) in the February 6, 1984 submittal. The Technical Specification page was revised as a part of the September 6, 1984, submittal. We find the change acceptable.

3.0 Environmental Consideration

An Environmental Assessment and Finding of No Significant Impact has been issued for this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 31, 1984

Principal Reviewers: R. Kendall, G. Schwenk

U.S. NUCLEAR REGULATORY COMMISSION

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

DOCKET NO. 50-321

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 105 to Facility Operating License No. DPR-57, issued to Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), which revised the Technical Specifications for operation of the Edwin I. Hatch Nuclear Plant, Unit No. 1, (the facility) located in Appling County, Georgia. The amendment was effective as of the date of its issuance.

This amendment revised the Technical Specifications to implement the Average Power Range Monitor/Rod Block Monitor/Technical Specification (ARTS) Improvement Program.

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 Consideration of Issuance of Amendment and Opportunity for Prior Hearing in connection with this action was published in the FEDERAL REGISTER on May 16, 1984, 48 FR 20769 and October 24, 1984, 48 FR 42819. No request for a hearing or petition for leave to intervene was filed following this notice.

Also, in connection with this action, the Commission prepared an Environmental Assessment and Finding of No Significant Impact which was published in the FEDERAL REGISTER on December 21, 1984, 49 FR 49741.

For further details with respect to this action, see (1) the application for amendment dated February 6, 1984, as supplemented September 6, 1984, (2) Amendment No. 105 to License No. DPR-57, and (3) the Commission's related Safety Evaluation. All of these items are 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, 301 City Hall Drive, Baxley, Georgia. 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 Licensing.

Dated at Bethesda, Maryland, this 31st day of December 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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