DOCKET

May 31, 1990

Docket No. 50-333

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Mr. John C. Brons
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 75871)

The Commission has issued the enclosed Amendment No. 162 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated January 12, 1990, which was amended and superseded by letter dated April 20, 1990.

The amendment removes the cycle-specific parameter limits from the Technical Specifications and places them in the Core Operating Limits Report in accordance with guidance contained in Generic Letter 88-16.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

David E. LaBarge, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.162 to DPR-59
2. Safety Evaluation

cc: w/enclosures
See next page

* See previous concurrence

PDI-1	PDI-1	SRXB*	SRXB*	OGC*	PDI-1
CVogan	DLaBarge:rsc	RJones	DFieno		RACapra
5/ /90	5/30 /90	5/ 8 /90	5/8 /90	5/23 /90	5/31 /90

DOCUMENT NAME: ISSUANCE OF AMENDMENT 75871

DF01
1/1

Mr. John C. Brons
Power Authority of the State of New York
James A. FitzPatrick Nuclear
Power Plant

cc:

Mr. Gerald C. Goldstein
Assistant General Counsel
Power Authority of the State
of New York
1633 Broadway
New York, New York 10019

Resident Inspector's Office
U. S. Nuclear Regulatory Commission
Post Office Box 136
Lycoming, New York 13093

Mr. William Fernandez
Resident Manager
James A. FitzPatrick Nuclear
Power Plant
Post Office Box 41
Lycoming, New York 13093

Mr. J. A. Gray, Jr.
Director Nuclear Licensing - BWR
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Supervisor
Town of Scriba
R. D. #4
Oswego, New York 13126

Mr. J. P. Bayne, President
Power Authority of the State
of New York
1633 Broadway
New York, New York 10019

Mr. Richard Patch
Quality Assurance Superintendent
James A. FitzPatrick Nuclear
Power Plant
Post Office Box 41
Lycoming, New York 13093

Charlie Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. A. Klausman
Senior Vice President - Appraisal
and Compliance Services
Power Authority of the State
of New York
1633 Broadway
New York, New York 10019

Mr. George Wilverding, Manager
Nuclear Safety Evaluation
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Mr. R. E. Beedle
Vice President Nuclear Support
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Mr. S. S. Zulla
Vice President Nuclear Engineering
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Mr. William Josiger, Vice President
Operations and Maintenance
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 162
License No. DPR-59

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated January 12, 1990, which was amended and superseded by letter dated April 20, 1990, 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-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.162 , 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 the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 31, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 162

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
vii	vii
6	6
6a	6a
8	8
8a	--
9	9
10	10
10a	--
12	12
13	13
14	14
15	15
16	16
18	18
20	20
30f	30f
31	31
31a	--
35	35
41	41
43	43
43a	--
47a	--
47b	--
47c	--
47d	--
72	72
73	73
74	74
102	102
123	123
124	124
130	130
135a	--
135b	--
135c	--
135d	--
135e	--
135f	--
135g	--
135h	--
135i	--
135j	--
135k	--
135l	--
245	245
254-c thru 254f	254-c
	254-d thru 254-f

LIST OF FIGURES

1.0 (cont'd)

surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted ±25 percent. The interval as pertaining to instrument and electric surveillance shall never exceed one operating cycle. In cases where the elapsed interval has exceeded 100 percent of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

U. Thermal Parameters

1. Minimum critical power ratio (MCPR)- Minimum value of the ratio of that power in a fuel assembly which is calculated to cause some point in that fuel assembly to experience boiling transition to the actual assembly operating power for all fuel assemblies in the core.

2. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR.

3. Maximum Fraction of Limiting Power Density - The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

4. Transition Boiling - Transition boiling means the boiling region between nucleate and film boiling. Transition boiling is the region in which both nucleate and film boiling occur intermittently with neither type being completely stable.

V. Electrically Disarmed Control Rod

To disarm a rod drive electrically, the 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. Electrical disarming does not eliminate position indication.

W. High Pressure Water Fire Protection System

The High Pressure Water Fire Protection System consists of: a water source and pumps; and distribution system piping with associated post indicator valves (isolation valves). Such valves include the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler or water spray subsystem.

X. Staggered Test Basis

A Staggered Test Basis shall consist of:

a. A test schedule for "n" systems, subsystems, trains or other designated components obtained by dividing the specified test interval into "n" equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

Y. Rated Recirculation Flow

That drive flow which produces a core flow of 77.0×10^6 lb/hr.

Z. Top of Active Fuel

The Top of Active Fuel, corresponding to the top of the enriched fuel column of each fuel bundle, is located 352.5 inches above vessel zero, which is the lowest point in the inside bottom of the reactor vessel. (See General Electric drawing No. 919D690BD.)

AA. Rod Density

Rod density is the number of control rod notches inserted expressed as a fraction of the total number of control rod notches. All rods fully inserted is a condition representing 100 percent rod density.

AB. Purge-Purging

Purge or Purging is the controlled process of discharging air or gas from a confinement in such a manner that replacement air or gas is required to purify the confinement.

AC. Venting

Venting is the controlled process of releasing air or gas from a confinement in such a manner that replacement air or gas is not provided or required.

AD. Core Operating Limits Report (COLR)

This report is the plant-specific document that provides the core operating limits for the current operating cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.A.4. Plant operation within these operating limits is addressed in individual Technical Specifications.

1.1 (cont'd)

B. Core Thermal Power Limit (Reactor Pressure ≤785 psig)

When the reactor pressure is ≤785 psig or core flow is less than or equal to 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 (cont'd)

A.

1.

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

APRM - The APRM flux scram setting shall be ≤15 percent of rated neutron flux with the Reactor Mode Switch in Startup/Hot Standby or Refuel.

c. APRM Flux Scram Trip Settings (Run Mode)

(1) Flow Referenced Neutron Flux Scram Trip Setting

When the Mode Switch is in the RUN position, the APRM flow referenced flux scram trip setting shall be less than or equal to the limit specified in Table 3.1-1. This setting shall be adjusted during single loop operation when required by Specification 3.5.J.

For no combination of recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 117% of rated thermal power.

1.1 (cont'd)

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

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 18 inches above the Top of Active Fuel when it is seated in the core.

2.1 (cont'd)

(2) Fixed High Neutron Flux Scram Trip Setting

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

S $\leq$ 120% Power

d. APRM Rod Block Setting

The APRM Rod block trip setting shall be less than or equal to the limit specified in Table 3.2-3. This setting shall be adjusted during single loop operation when required by Specification 3.5.J.

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1.1 BASES

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 minimum critical power ratio (MCPR). This Safety Limit represents a conservative margin relative to the 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 conditions represent a significant departure from the condition intended by design for planned operation.

A. Reactor Pressure >785 psig 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 would 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 critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variable, i.e., the operating domain. The current load line limit analysis contains the current operating domain map. The Safety Limit MCPR has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the MCPR operating limit in the Core Operating Limits Report, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The MCPR fuel cladding safety limit is increased by 0.01 for single-loop operation as discussed in Reference 2. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including the uncertainty in the boiling transition correlation as described in Reference 1. The uncertanties employed in deriving the Safety Limit are

1.1 (cont'd)

provided in Reference 1. Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of fuel assembly at the Safety Limit would not produce boiling transition. Thus, although it is not required to establish the safety limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

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 FitzPatrick 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 operation (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 (Safety Limit), operation is constrained by the maximum LHGR identified in the Core Operating Limits Report.

At 100% power, this limit is reached with maximum fraction of limiting power density (MFLPD) equal to 1.00. In the event of operation with MFLPD greater than the fraction of rated power (FRP), the APRM scram and rod block settings shall be adjusted as specified in Tables 3.1-1 and 3.2-3 respectively.

B. Core Thermal Power Limit (Reactor Pressure <785 psig)

At pressures below 785 psig the core elevation pressure drop is greater than 4.56 psi for no boiling in the bypass region. At low powers and flows, this pressure drop is due to the elevation pressure of the bypass region of the core. Analysis shows that for bundle power in the range of 1-5 MWt, the channel flow will never go below 28 x 10^3 lb/hr. This flow results from the pressure differential between the bypass region and the fuel channel. The pressure differential is primarily a result of changes in the elevation pressure drop due to the density difference between the boiling water in the fuel channel and the non-boiling water in the bypass region. Full scale ATLAS test data taken at pressures from 0 to 785 psig indicate that the fuel assembly critical power at 28 x 10^3 lb/hr 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 785 psig is conservative.

1.1 BASES (Cont'd)

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 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.

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

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the Safety Limit at 18 in. above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

E. References

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P, latest approved revision and amendments.

2. FitzPatrick Nuclear Power Plant Single-Loop Operation, NEDO 24281, August 1980.

BASES

2.1 FUEL CLADDING INTEGRITY

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

The transient analyses performed for each reload are given in Reference 2. Models and model conservatism are also described in this reference. As discussed in Reference 4, the core wide transient analysis for one recirculation pump operation is conservatively bounded by two-loop operation analysis, and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation.

Fuel cladding integrity is assured by the applicable operating limit MCPR for steady state conditions given in the Core Operating Limits Report (COLR). These operating limit MCPR's are derived from the established fuel cladding integrity Safety Limit, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient.

The most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO. The type of transients evaluated were increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit, the required operating limit MCPR in the Core Operating Limits Report is obtained.

The evaluation of a given transient begins with the system initial parameters shown in the current reload analysis and Reference 2 that are input to the core dynamic behavior transient computer programs described in Reference 2. The output of these programs along with the initial MCPR form the input for the further analyses of the thermally limited bundle with a single channel transient thermal hydraulic code. The principal result of the evaluation is the reduction in MCPR caused by the transient.

2.1 BASES (Cont'd)

The MCPR operating limits in the COLR are 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 is not permitted. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

- The abnormal operational transients were analyzed to the licensed maximum power level.
- The licensed maximum power level is 2436 MWt.
- Analyses of transients employ adequately conservative values of the controlling reactor parameters.
- 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 accommodate 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.

2.1 BASES (Cont'd)

c. APRM Flux Scram Trip Setting (Run Mode) (cont'd)

rated power. This reduced flow referenced trip setpoint will result in an earlier scram during slow thermal transients, such as the loss of 80°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 neutron flux signal 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.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted as specified in Table 3.1-1 when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by either (1) reducing the APRM scram and rod block settings or (2) adjusting the indicated APRM signal to reflect the high peaking condition.

Analyses of the limiting transients show that no scram adjustment is required to assure that the MCPR will be greater than the Safety Limit when the transient is initiated from the MCPR operating limits specified in the Core Operating Limits Report.

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 provides an added level of protection before APRM Scram. 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 withdrawal. The flow variable trip setting parallels that of the APRM Scram and provides margin to scram, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. 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 fraction of limiting power density exceeds the fraction of rated power, thus preserving the APRM rod block margin. As with the scram setting, this may be accomplished by adjusting the APRM gain.

2. Reactor Water Low Level Scram Trip Setting

The reactor low water level scram is set at a point which will assure that the water level used in the Bases for the Safety Limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

2.1 BASES (Cont'd)

C. References

1. (Deleted)
2. "General Electric Standard Application for Reactor Fuel", NEDE 24011-P-A (Approved revision number applicable at time that reload fuel analyses are performed).
3. (Deleted)
4. FitzPatrick Nuclear Power Plant Single-Loop Operation, NEDO-24281, August, 1980.

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate the reactor scram.

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. The setpoints, minimum number of trip systems, minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as shown on Table 3.1-1. The design system response time from the opening of the sensor contact to and including the opening of the trip actuator contacts shall not exceed 50 msec.

B. Minimum Critical Power Ratio (MCPR)

During reactor power operation, the MCPR operating limit shall not be less than that specified in the Core Operating Limits Report.

1. During Reactor power operation with core flow less than 100% of rated, the MCPR operating limit shall be multiplied by the appropriate K_f as specified in the Core Operating Limits Report.

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type of frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1-1 and 4.1-2 respectively.

B. Maximum Fraction of Limiting Power Density (MFLPD)

The MFLPD shall be determined daily during reactor power operation at $\geq$25% rated thermal power and the APRM high flux scram and Rod Block trip settings adjusted if necessary as specified in the Core Operating Limits Report.

3.1 (cont'd)

2. If anytime during reactor operation at greater than 25% of rated power it is determined that the operating limit MCPR is being exceeded, action shall then be initiated within fifteen (15) minutes to restore operation to within the prescribed limits. If the MCPR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall begin immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the MCPR is returned to within the prescribed limits.

4.1 (cont'd)

C. MCPR shall be determined daily during reactor power operation at $\geq$25% of 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.B.5.

D. 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.

E. Verification of the MCPR operating limits shall be performed as specified in the Core Operating Limits Report.

3.1 BASES (cont'd)

Turbine control valves fast closure initiates a scram based on pressure switches sensing electro-hydraulic control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative (500<P<850 psig) to the normal (EHC) oil pressure of 1,600 psig so that based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure.

The requirement that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the start-up and refuel modes assures that there is proper overlap in the neutron monitoring system functions and thus, that adequate coverage is provided for all ranges of reactor operation.

B. The limiting transient which determines the required steady state MCPR limit depends on cycle exposure. The operating limit MCPR values as determined from the transient analysis in the current reload submittal for various core exposures are specified in the Core Operating Limits Report (COLR).

The ECCS performance analyses assumed reactor operation will be limited to MCPR = 1.20, as described in NEDO-21662 and NEDC-31317P. The Technical Specifications limit operation of the reactor to the more conservative MCPR based on consideration of the limiting transient as specified in the COLR.

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Mode in Which Function Must be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6)(16)	Startup	Run		
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 Selections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
3	IRM High Flux	≤ 120/125 of full scale	X	X		8 Instrument Channels	A
3	IRM Inoperative		X	X		8 Instrument Channels	A
2	APRM Neutron Flux-Startup (15)	≤ 15% Power	X	X		6 Instrument Channels	A
2	APRM Flow Referenced Neutron Flux (Not to exceed 117%) (13)(14)	(12)			X	6 Instrument Channels	A or B
2	APRM Fixed High Neutron Flux (14)	≤ 120% Power			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B

Amendment No. ~~18~~, ~~30~~, ~~48~~, ~~72~~, ~~87~~, ~~98~~, ~~124~~ 162

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1- (cont'd)

C. High Flux IRM.

D. Scram Discharge Volume High Level when any control rod in a control cell containing fuel is not fully inserted.

E. APRM 15% Power Trip.

7. Not required to be operable when primary containment integrity is not required.
8. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.
9. The APRM downscale trip is automatically bypassed when the IRM Instrumentation is operable and not high.
10. An APRM will be considered operable if there are at least 2 LPRM inputs per level and at least 11 LPRM inputs of the normal complement.
11. See Section 2.1.A.1.
12. The APRM Flow Referenced Neutron Flux Scram setting shall be less than or equal to the limit specified in the Core Operating Limits Report.
13. The Average Power Range Monitor scram function is varied as a function of recirculation flow (W). The trip setting of this function must be maintained as specified in the Core Operating Limits Report.
14. The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.
15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is place in the Run position.
16. *During the proposed Hydrogen Addition Test, the background radiation level will increase by approximately a factor of 5 for peak hydrogen concentration. Therefore, within 24 hours prior to performance of the test, the Main Steam Line Radiation Monitor Trip Level Setpoint will be raised to $\leq$ three times the anticipated radiation levels. Upon completion of the Hydrogen Addition Test, the setpoint will be readjusted to its prior setting within 24 hours.

* This specification is in effect only during Operating Cycle 10.

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

Minimum No. of Operable Instrument Channels Per Trip System	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Action
2	APRM Upscale (Flow Biased)	(8)	6 Inst. Channels	(1)
2	APRM Upscale (Start-up Mode)	$\leq$ 12%	6 Inst. Channels	(1)
2	APRM Downscale	$\geq$ 2.5 indicated on scale	6 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Flow Biased)	(8)	2 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Downscale)	$\geq$ 2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (2)	$\geq$ 2% of full scale	8 Inst. Channels	(1)
3	IRM Detector not in Start-up Position	(7)	8 Inst. Channels	(1)
3	IRM Upscale	$\leq$ 86.4% of full scale	8 Inst. Channels	(1)
2 (4)	SRM Detector not in Start-up Position	(3)	4 Inst. Channels	(1)
2 (4)(5)	SRM Upscale	$\leq 10^5$ counts/sec	4 Inst. Channels	(1)
1	Scram Discharge Instrument Volume High Water Level	$\leq$ 26.0 gallons per instrument volume	2 Inst. Channels	(9)(10)

NOTES FOR TABLE 3.2-3

1. For the Start-up and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function. The SRM and IRM block need not be operable in run mode, and

TABLE 3.2-3 (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2.-3

the RBM rod block need not be operable in start-up mode. When the reactor is in the start-up mode, the APRM upscale (start-up mode) rod block shall be operable. When the reactor is in the run mode, the APRM upscale (flow biased) and APRM downscale rod blocks shall be operable. From and after the time it is found that the first column cannot be met for one of the two trip systems, this 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. From and after the time it is found that the first column cannot be met for both trip systems, the systems shall be tripped.

2. IRM downscale is bypassed when it is on its lowest range.
3. This function is bypassed when the count rate is $\geq$ 100 cps.
4. One of the four SRM inputs may be bypassed.
5. This SRM Function is bypassed when the IRM range switches are on range 8 or above.
6. The trip is bypassed when the reactor power is $\leq$ 30%.
7. This function is bypassed when the Mode Switch is placed in Run.
8. The Flow Biased APRM Upscale and Rod Block Monitor trip level setpoint shall be less than or equal to the limit specified in the Core Operating Limits Report.
9. When the reactor is subcritical and the reactor water temperature is less than 212°F, the control rod block is required to be operable only if any control rod in a control cell containing fuel is not fully inserted.
10. When one of the instruments associated with scram discharge instrument volume high water rod blocks is not operable, the trip system shall be tripped.

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3.3 and 4.3 BASES (cont'd)

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage.

   This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (e.g., MCPR limit). During use 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 withdraw does not occur. It is 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.

C. Scram Insertion Times

The Control Rod System is designated to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit. Scram insertion time test criteria of Section 3.3.C.1 were used to generate the generic scram reactivity curve shown in NEDE-24011-P-A. This generic curve was used in analysis of non-pressurization transients to determine MCPR limits. Therefore, the required protection is provided.

The numerical values assigned to the specified scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on JAFNPP.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes of satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types. These values are specified in the Core Operating Limits Report. If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then 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, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

4.5 (cont'd)

2. Following any period where the LPCI subsystems or core spray subsystems have not been maintained in a filled condition; the discharge piping of the affected subsystem shall be vented from the high point of the system and water flow observed.

3. Whenever the HPCI or RCIC System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI or RCIC shall be vented from the high point of the system, and water flow observed on a monthly basis.

4. The level switches located on the Core Spray and RHR System discharge piping high points which monitor these lines to insure they are full shall be functionally tested each month.

H. 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.

3.5 (cont'd)

I. Linear Heat Generation Rate (LHGR)

The linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR specified in the Core Operating Limits Report.

If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for LHGR is being exceeded, action shall then 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, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the LHGR is returned to within the prescribed limits.

4.5 (cont'd)

I. Linear Heat Generation Rate (LHGR)

The LHGR shall be determined daily during reactor operation at $\geq$25% rated thermal power.

3.5 BASES (cont'd)

requirements for the emergency diesel generators.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, RCIC, and HPCI are not filled, a water hammer can develop in this piping when the pump(s) are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this technical specification requires the discharge lines to be filled whenever the system is required to be operable. If a discharge pipe is not filled, the pumps the supply that line must be assumed to be inoperable for technical specification purposes. However, if a water hammer were to occur, the system would still perform its design function.

H. 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 10 CFR 50 Appendix K.

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 are within the 10 CFR 50 Appendix K limit. The limiting values for APLHGR are specified in the Core Operating Limits Report. During Single Loop Operation a multiplier is applied to these values. The derivation of this multiplier can be found in Bases 3.5.K, Reference 1.

I. 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.

The LHGR shall be checked daily during reactor operation at 25% rated thermal 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 ratio of local LHGR to average LHGR would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

5.0 DESIGN FEATURES

5.1 SITE

A. The James A. FitzPatrick Nuclear Power Plant is located on the PASNY portion of the Nine Mile Point site, approximately 3,000 ft. east of the Nine Mile Point Nuclear Station, Unit 1. The NPP-JAF site is on Lake Ontario in Oswego Country, New York, approximately 7 miles northeast of Oswego. The plant is located at coordinates north 4,819, 545.012 m, east 386, 968.945 m, on the Universal Transverse Mercator System.

B. The nearest point on the property line from the reactor building and any points of potential gaseous effluents, with the exception of the lake shoreline, is located at the northeast corner of the property. This distance is approximately 3,200 ft. and is the radius of the exclusion areas as defined in 10 CFR 100.3.

5.2 REACTOR

A. The reactor core consists of not more than 560 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of slightly enriched uranium dioxide (UO_2) as fuel material. Fuel assemblies shall be limited to those fuel designs approved by the NRC staff for use in BWRs.

B. The reactor core contains 137 cruciform-shaped control rods as described in Section 3.4 of the FSAR.

5.3 REACTOR PRESSURE VESSEL

The reactor pressure vessel is as described in Table 4.2-1 and 4.2-2 of the FSAR. The applicable design codes are described in Section 4.2 of the FSAR.

5.4 CONTAINMENT

A. The principal design parameters and characteristics for the primary containment are given in Table 5.2-1 of the FSAR.

B. The secondary containment is as described in Section 5.3 and the applicable codes are as described in Section 12.4 of the FSAR.

C. Penetrations of the primary containment and piping passing through such penetrations are designed in accordance with standards set forth in Section 5.2 of the FSAR.

5.5 FUEL STORAGE

A. The new fuel storage facility design criteria are to maintain a K_{eff} dry <0.90 and flooded <0.95. Compliance shall be verified prior to introduction of any new fuel design to this facility.

(A) ROUTINE REPORTS (Continued)

4. CORE OPERATING LIMITS REPORT

a. Core operating limits shall be established prior to startup from each reload cycle, or prior to any remaining portion of a reload cycle for the following:

- The Average Planar Linear Heat Generation Rates (APLHGR) of Specification 3.5.H;
- The Minimum Critical Power Ratio (MCPR) and MCPR low flow adjustment factor, K_f, of Specifications 3.1.B and 4.1.E;
- The Linear Heat Generation Rate (LHGR) of Specification 3.5.I;
- The Reactor Protection System (RPS) APRM flow biased trip settings of Table 3.1-1; and
- The flow biased APRM and Rod Block Monitor (RBM) rod block settings of Table 3.2-3.

and shall be documented in the Core Operating Limits Report (COLR).

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC as described in:

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P, latest approved version and amendments.
2. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, October, 1986 including latest errata and addenda.
3. "Loss-of-Coolant Accident Analysis for James A. FitzPatrick Nuclear Power Plant," NEDO-21662-2, July, 1977 including latest errata and addenda.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements thereto, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

INTRODUCTION

By letter dated January 12, 1990, as amended and superseded by letter dated April 20, 1990, the Power Authority of the State of New York (the licensee), proposed changes to the Technical Specifications (TS) for the James A. FitzPatrick Nuclear Power Plant. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) which contains the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements in the Administrative Controls section of the TS. Guidance on the proposed changes was developed by the NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988.

EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

1. The Definition section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with NRC-approved methodologies that maintains the limit of the safety analysis. The definition states that plant operation within these limits is addressed in individual Technical Specifications.

2. The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

   a. Specification 3.5.H

      The Average Planar Linear Heat Generation Rate (APLHGR) limits for this specification are specified in the COLR.

b. Specifications 3.1.B and 4.1.E

The Minimum Critical Power Ratio (MCPR) limits and the MCPR flow adjustment factor K_f for these specifications are specified in the COLR.

c. Specification 3.5.I

The Linear Heat Generation Rate (LHGR) limits for this specification are specified in the COLR.

d. Specification Table 3.1-1

The Reactor Protection System (RPS) flow biased trip settings of Technical Specification Table 3.1-1 are specified in the COLR.

e. Specification Table 3.2-3

The Control Rod Block flow biased APRM and Rod Block Monitor (RBM) rod block settings of Technical Specification Table 3.2-3 are specified in the COLR.

The changes to the specifications also required changes to the Bases to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these Bases are acceptable.

3. Specification 6.9.A.4 was added to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

a. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P, latest approved version and amendment.

b. "James A. FitzPatrick Nuclear Power Plant SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," NEDC-31317P, October 1986 including latest errata and addenda.

c. "Loss of Coolant Accident Analysis for James A. FitzPatrick Nuclear Power Plant," NEDO-21662-2, July 1977 including latest errata and addenda.

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to the NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the value of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. As a result of this review and the amended submittal, the staff recommended a number of changes to the draft COLR. In particular, references to design feature of the fuel assemblies will not be include in the COLR. The licensee agreed with the suggested changes. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

The following additional changes have also been proposed by the licensee in this submittal:

1. Specification 2.1.A.1.c.(1) would be modified to remove the APRM high neutron flux scram trip setting formulas (Run Mode) and replaced them with reference to Table 3.1-1 and Specification 3.5.J which will contain the appropriate limits in accordance with this amendment.

2. Specification 2.1.A.1.d would be modified to remove the APRM rod block trip setting formulas (Run Mode) and replace them with reference to Table 3.2-3 and Specification 3.5.J which will contain the appropriate limits in accordance with this amendment.

3. The value of the MCPR safety limit (1.04) quoted in Bases 1.1 would be removed and the words "Safety Limit" substituted to clarify the meaning of the terminology. Other non-technical, administrative changes were also proposed to this Bases section and Bases Section 2.1.

4. Specification 5.2.A would be modified to remove the specific fuel types from the Reactor Design Features section and insert a more generalized statement which describes the fuel assemblies composition and that they are composed of fuel designs approved by the NRC staff for use in BWRs.

The above changes have been proposed to better consolidate the various limits and information. They are administrative in nature and, therefore, are acceptable.

Another proposed change would modify Section 2.1 Bases to state that transient analyses for Abnormal Operational Transients are performed at the nominal 100 percent power (2436 MWt) rather than the maximum power level of 2535 MWt (corresponding to 104 percent power). This method of analysis is based on GEMINI methods and was previously approved in Amendment No. 109. This change is acceptable.

In addition, the licensee has proposed removal of a number of pages that are now labeled as blank or from which the specifications are being moved to other page and will, therefore, become blank. These changes are also administrative and are acceptable.

SUMMARY

We have reviewed the request by the Power Authority of the State of New York to modify the Technical Specifications of the James A. FitzPatrick Nuclear Power Plant that would remove the specific values of some cycle-dependent parameters from the specifications and place the values in a Core Operating Limits Report that would be referenced by the specifications. Based on this review, we conclude that these Technical Specification modifications are acceptable because they are in accordance with Generic Letter 88-16. We have also reviewed the changes to Specification 5.2.A on design features of the reactor core and fuel assemblies and other administrative changes and conclude that they are acceptable.

EXIGENT CIRCUMSTANCES

The Commission's regulations, 10 CFR 50.91, contain provisions for issuance of amendments when the usual 30-day public notice period cannot be met. One type of special exception is an exigency. An exigency is a case where the staff and licensee need to act promptly, but failure to act promptly does not involve a plant shutdown, derating, or delay in startup. The exigency case usually represents an amendment involving a safety enhancement to the plant.

Under such circumstances, the Commission notifies the public in one of two ways: by issuing a Federal Register notice providing an opportunity for hearing and allowing at least two weeks for prior public comments, or by issuing a press release discussing the proposed changes, using the local media. In this case, the Commission used the first approach.

The licensee submitted the request for amendment on January 12, 1990 to incorporate changes to the Technical Specifications to remove cycle-specific parameters in accordance with Generic Letter 88-16. The licensee requested that the amendment be issued prior to May 15, 1990, at which time the plant was expected to startup from the 1990 refueling outage.
It was noticed in the Federal Register on March 27, 1990 (55 FR 8234), at which time the staff proposed a no significant hazards consideration determination.

Following discussions with the staff which clarified implementation details of the generic letter, the licensee superseded the original amendment by letter dated April 20, 1990. The only technical change to the original submittal involved relocation of the Fuel Design Features, which lists the different fuel assemblies by coded designators, from the Technical Specifications to the Core Operating Limits Report (COLR). Since this change was not in accordance with the present staff interpretation of the generic letter, the licensee deleted it from the new amendment application. Since this represented a significant change from what was previously noticed, the change was noticed in the Federal Register on April 30, 1990 (55 FR 18042). In this notice the staff proposed to determine that the amended application involved no significant hazards consideration and offered a 15 day comment period in order to enable issuance of the amendment in accordance with the licensee's expected startup date.

The net effect of the change is a more restrictive set of Technical Specifications which state that the fuel assemblies shall be limited to those fuel designs approved by the NRC staff for use in boiling water reactors.

Therefore, the staff is issuing the amendment under exigent circumstances. The licensee did not request emergency treatment of the amended application and the staff does not believe that an emergency situation exists. However, the staff does believe that the amendment should be issued prior to plant startup from the present refueling outage.

There were no public comments in response to the either notices published in the Federal Register.

FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commissions regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

Operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes which address Generic Letter 88-16 merely move cycle-specific parameter limits from the Technical Specifications to the Core Operating Limits Report. NRC-approved methodologies will continue to be used as the basis for establishing the limits and incorporting the values into the Core Operating Limits Report, thereby ensuring that the proper values are used. The submittal of this document to the NRC will allow the staff to continue to monitor the values and process. The proposed change to the Bases of Section 2.1 (the use of 100 percent power in the analysis of abnormal operational transients using GEMINI methods rather that 104 percent power) has been previously reviewed and approved by the NRC for both generic and FitzPatrick application. It showed that power level measurement uncertainties are accounted for adequately in the Minimum Critical

Power Ratio (MCPR) Operating Limit. The level of confidence that this safety limit will not be violated as a result of a transient is not reduced. Other proposed changes are administrative in nature and serve to clarify terminology.

Operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. No safety-related equipment, function, or plant operation will be altered as a result of the proposed changes and they do not create any new accident mode. The limits will continue to be in effect and updated as required. The level of document control and quality assurance applied by the licensee to the preparation and use of changes to the Core Operating Limits Report will be equivalent to that applied to the Technical Specification changes. In addition, the MCPR operating limit criteria of Bases Section 2.1 continues to be determined using approved methodology. Other proposed changes are administrative in nature and serve to clarify terminology.

Operation of the facility in accordance with the proposed amendment will not involve a significant reduction in a margin of safety. The Generic Letter 88-16 changes are administrative in nature and involve moving limits from one document to another. They do not impact plant operation. The proposed changes still require operation within the limits determined using NRC-approved methods and appropriate remedial actions be taken if the limits are violated. For the changes to Bases Section 2.1, the MCPR operating limit continues to be determined using an approved methodology that conservatively accounts for power level measurement uncertainties. The same criteria for acceptable operation is maintained. Other proposed changes are administrative in nature and serve to clarify terminology.

Based upon the above considerations, the staff concludes that the amendment meets the three criteria of 10 CFR 50.92. Therefore, the staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes recordkeeping or reporting requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. In addition, the Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration based on the original submittal and there has been no public comment on such finding. Also, the Commission has made a final consideration that the amendment does not involve a signficiant hazards consideration in this document. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9) and 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

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: May 31, 1990

PRINCIPAL CONTRIBUTORS:

D. Fieno
D. LaBarge