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Docket No. 50-333

Power Authority of the State of New York
ATTN: Mr. George T. Berry
General Manager and
Chief Engineer
10 Columbus Circle
New York, New York 10019

Gentlemen:

The Commission has issued the enclosed Amendment No. 18 to Facility Operating License No. DPR-59 for the FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your application for amendment submitted by letter dated May 5, 1976.

The amendment provides for a reduction in the operating minimum critical power ratio consistent with low core exposure while preserving the existing safety margin.

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

Sincerely,

Original signed by
Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors

Enclosures:
1. Amendment No. 18
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

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

May 21, 1976

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

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 18
License No. DPR-59

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

   A. The application for amendment by the Power Authority of the State of New York and Niagara Mohawk Power Corporation (the licensees) sworn to May 4, 1976, 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. After weighing the environmental aspects involved, 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 a change to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance:
May 21, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 18

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace pages 12, 15, 16, 18, 29, 30, 31, 35, 40, 71, 72, 73, 74, 102 and 103 of the Appendix A Technical Specifications with the attached pages bearing the same numbers. Changes on these pages are shown by marginal lines. Pages 71 and 74 are unchanged and are included for convenience only.

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 such that the minimum critical power ratio (MCPR) is no less than 1.06. MCPR > 1.06 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 variables, i.e., normal plant operation presented on Figure 1.1-1 by the nominal expected flow control line. The Safety Limit (MCPR of 1.06) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition (MCPR > 1.22 for cycle-1 exposures up to 8500 MWD/T and > 1.35 from 8500 MWD/T to end of cycle conditions) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit 1.06 is derived from a detailed statistical analysis considering all of the

2.1 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 2535 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7-1 of the FSAR. In addition, 2436 is the licensed maximum power level of Fitzpatrick, 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 five dollars of negative reactivity at End of Cycle (or approximately 16 dollars at 8500 MWD/T) have been inserted 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.

2.1 BASES *(Cont'd.)*

For analyses of the thermal consequences of the transients a MCPR of > 1.22 for cycle-1 exposures up to 8500 MWD/T and > 1.35 from 8500 MWD/T to end of cycle-1 conditions 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 power and flow relationships has considered operation with either one or two recirculation pumps.

In summary:

i. *The abnormal operational transients were analyzed to a power level of 2535 MWt.*

ii. *The licensed maximum power level is 2436 MWt.*

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

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

A. *Trip Settings*

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

1. *Neutron Flux Trip Settings*

a. *IRM Flux Scram Trip Setting*

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

2.1 BASES (Cont'd.)

system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnromal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin. An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.A.1.c, when the maximum total peaking factor is greater than 2.60.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.06 when the transient is initiated from MCPR > 1.22 for cycle-1 exposures up to 8500 MWD/T and > 1.35 from 8500 MWD/T to end of cycle-1 conditions.

d. APRM Rod Block Trip Setting

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

1.2 and 2.2 BASES

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575° F for the reactor vessel, 1148 psig at 568°F for the recirculation suction piping and 1274 psig at 575° F for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% X 1,250 = 1,375 psig), and the ANSI Code permits pressure transients up to 20 percent over the design pressure (120% X 1,150 = 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

The analysis in NEDO-21166-1, Section 7.2.3 shows that the main steam isolation valve transient, when direct scram is ignored, is the most severe event resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is 105 psig above the peak pressure produced by the event above. Thus, the pressure safety limit is well above the peak pressure that can result from reasonably expected (1,375 psig) overpressure transients. Figure 7-4 of NEDO-21166-1 presents the curve produced by this analyses. Reactor pressure is continuously indicated in the control room during operation.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

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 100 msec.*

B. Minimum Critical Power Ratio (MCPR)

MCPR shall be > 1.22 at rated power and flow for cycle-1 exposures from up to 8500 MWD/T and > 1.35 at rated power and flow from 8500 MWD/T to end of cycle-1 conditions. If at any time during steady state operation it is determined that the limiting value for MCPR is being exceeded action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the

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 and 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. *Daily, during reactor power operation, while in the RUN MODE, the peak heat flux and peaking factor shall be checked and the SCRAM and APRM Rod Block settings given by equations in Specifications 2.1.A.1 and 2.1.B shall be calculated if the peaking factor exceeds 2.6.*

3.1 (cont'd)

reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. For core flows other than rated, the MCPR shall be > 1.22 for cycle-1 exposures up to 8500 MWD/T and > 1.35 from 8500 MWD/T to end of cycle-1 conditions times K_f where K_f is as shown in Figure 3.1.1.

C. MCPR shall be determined 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.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.

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 startup 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. At cycle-1 exposure up to 8500 MWD/T it is the rod withdrawal error transient. It yields the largest ΔMCPR (0.16) which when added to the Safety Limit MCPR of 1.06 yields the minimum operating limit of 1.22. At exposures from 8500 MWD/T to EOC-1 conditions, the turbine trip without bypass is limiting. The ΔMCPR is 0.29 and the operating limit MCPR is 1.35. The ECCS performance analysis assumed reactor operation will be limited to MCPR of 1.18. However, the Technical Specifications limit operation of the reactor to the more conservative MCPR based on consideration of the limiting transient as given above.

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TABLE 3.2-2 (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

NOTES FOR TABLE 3.2-2

1. Whenever any ECCS subsystem is required by specification 3.5 to be operable, there shall be two operable trip systems. From and after the time it is found that the first column cannot be met for one of the trip systems, that trip system shall be placed in the tripped condition or the reactor shall be placed in the cold condition within 24 hours.

2. Instrument set point corresponds to 18 in. above the top of active fuel.

3. HPCI has only one trip system for these sensors.

4. Refer to Table 4.2-2, item 6.

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)	$S \leq (2.60/PF)S$	6 Inst. Channels	(1)
2	APRM Upscale (Start-up Mode)	$\leq 12\%$	6 Inst. Channels	(1)
2	APRM Downscale	≥ 2.5 indicated on scale	6 Inst. Channels	(1)
1 (6)	Rod Block Monitor (Flow Biased)	$S \leq 0.66W+40\%$ (8)	2 Inst. Channels	(1)
1 (6)	Rod Block Monitor Downscale	≥ 2.5 indicated on scale	2 Inst. Channels	(1)
3	IRM Downscale (2)	≥ 2.5 indicated on scale	8 Inst. Channels	(1)
3	IRM Detector not in Startup Position	(7)	8 Inst. Channels	(1)
3	IRM Upscale	≤ 108 indicated on scale	8 Inst. Channels	(1)
2 (4)	SRM Detector not in Startup Position	(3)	4 Inst. Channels	(1)
2 (4) (5)	SRM Upscale	$\leq 10^5$ counts/sec	4 Inst. Channels	(1)

NOTES FOR TABLE 3.2-3

1. For the Startup 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 blocks need not be operable in run mode, and the APRM and RBM rod blocks need not be operable in startup mode. From and after the time it is found that the first column cannot be met for one of the 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.

TABLE 3.2-3 (Cont'd)

INSTRUMENTATION THAT INITIATES CONTROL ROD BLOCKS

NOTES FOR TABLE 3.2-3 (Cont'd)

2. IRM downscale is bypassed when it is on its lowest range.
3. This function is bypassed when the count rate is ≥ 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 ≤ 30%.
7. This function is bypassed when the Mode Switch is placed in Run.
8. S = Rod Block Monitor setting in percent of initial

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

TABLE 3.2-4

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

Minimum No. of Operable Instrument Channels (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Action (2)
1	Refuel Area Exhaust Monitor	≤900 cpm	2 Inst. Channels	A or B
1	Reactor Building Area Exhaust Monitors	≤900 cpm	2 Inst. Channels	B
1	Off-gas Radiation Monitors	$\leq 7\times10^4$ mR/hr (3)	2 Inst. Channels	C
1	Turbine Bldg. Exhaust Monitors	≤900 cpm	2 Inst. Channels	C
1	Radwaste Bldg. Exhaust Monitor	≤900 cpm	2 Inst. Channels	C
1	Main Control Room Ventilation Monitor	$\leq 4\times10^3$ cpm	1 Inst. Channels	D
2	Mechanical Vacuum Pump Isolation	≤3 times normal full power background	4 Inst. Channels	E
1	Liquid Radwaste Discharge Monitor	(4)	1 Inst. Channel	F

NOTES FOR TABLE 3.2-4

1. Whenever the systems are required to be operable, there shall be two operable or tripped instrument channels per trip system. From and after the time it is found that this cannot be met, the indicated action shall be taken.

2. Action

   A. Cease operation of the refueling equipment.
   B. Isolate secondary containment and start the Standby Gas Treatment System.
   C. Refer to Section 2.3.B.4 of Environmental Technical Specification.
   D. Control Room isolation is manually initiated.
   E. Uses same sensors as Primary Containment Isolation on high main steam line radiation. Table 3.2-1.
   F. Refer to Environmental Technical Specification 2.3.A.3.

3. Refer to Specification 2.3.B of the Environmental Technical Specifications.

4. Trip setting to correspond to Specification 2.3.A of the Environmental Technical Specifications.

any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transient cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

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 (i.e., MCPR 1.22 for cycle-1 exposures up to 8500 MWD/T and 1.35 from 8500 MWD/T to end of cycle-1 conditions or LHGR = 18.5 kW/ft). 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 withdrawal does not occur. It is the responsibility of the Reactor Analyst 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. Other personnel qualified to perform this function may be designated by the Plant Superintendent

C. Scram Insertion Times

The Control Rod System is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage: i.e., to prevent the MCPR from becoming less than 1.06. The limiting power transient is that

resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (NEDO-21166-1, Figure 7-1) with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains greater than 1.06.

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.

In the analytical treatment of the transients, 390 msec are allowed between a neutron sensor reaching the scram point and the start of negative reactivity insertion. This is adequate and conservative when compared to the typically observed time delay of about 270 msec. Approximately 70 msec after neutron flux reaches the trip point, the pilot scram valve solenoid power supply voltage goes to zero and approximately 200 msec later, control rod motion begins. The 200 msec are included in the allowable scram insertion times specified in Specification 3.3.C.

The scram times generated at each refueling outage and during operation when compared to scram times generated during pre-operational tests demonstrate that the control rod drive scram function has not deteriorated. In addition, each instant when control rods are scram timed during operation or reactor trips, individual evaluations shall be performed to insure that control rod scram times have not deteriorated.

D. Reactivity Anomalies

During each fuel cycle, excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 18 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

INTRODUCTION

By an application for amendment to Operating License, submitted by letter dated May 5, 1976, the Power Authority of the State of New York and Niagara Mohawk Power Corporation (the licensees), proposed changes to the Technical Specifications appended to Facility Operating License No. DPR-59, for the James A. FitzPatrick Nuclear Power Plant. The proposed changes provide for the reduction of the operating limit minimum critical power ratio (MCPR) from 1.37 to 1.22.

BACKGROUND

We previously issued a Safety Evaluation (1) supporting a license amendment wherein we approved a minimum critical power ratio (MCPR), for operation, of 1.37 and a Safety Limit MCPR of 1.06.

At a MCPR of 1.0 a fuel assembly, at some point, will experience boiling transition. Boiling transition represents an unstable heat transfer condition on the fuel clad surface and could result in fuel clad temperatures exceeding allowable limits. The operating limit MCPR is derived by adding the effect of decrease in MCPR due to the most limiting operational transient (ΔMCPR) to the Safety Limit MCPR. In reference (1) a ΔMCPR of 0.31 was derived for end-of-cycle (EOC) core conditions; hence 1.06 + 0.31 = 1.37. Our analysis was based on a General Electric Report, NEDO-21166 (2). Reference (2) was based on EOC conditions and resulted in the imposition of requirements consistent with EOC conditions throughout the entire first cycle. The licensees, as part of the May 5, 1976 proposed license amendment have submitted an amended General Electric Report, NEDO-21166-1 (3) with operating MCPR

limits derived at a cycle exposure of 8500 MWD/T. At the present time (May 1976) the core has experienced about 2800 MWD/T of fuel exposure. Because ΔMCPR increases with core exposure, the proposed MCPR limits would be applicable from the present time until the core reached an exposure of 8500 MWD/T. Reference (3) and the proposed Technical Specification changes would allow the plant to reduce the operating limit MCPR to 1.22, while maintaining the same Safety Limit MCPR of 1.06. Thus ΔMCPR is reduced from 0.31 to 0.16. The reduction is justified in Reference (3) and in the Bases Sections of the Proposed Technical Specifications by a reanalysis of transients at a fuel exposure of 8500 MWD/T.

The previously evaluated and accepted nuclear considerations [See Nuclear Design in Reference (1)] remain unchanged by the reduction in operating limit MCPR.

EVALUATION

Operating Limit MCPR

Various transient events will reduce the operating CPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not exceeded during anticipated abnormal operational transients, the most limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio. The licensees have submitted the results of the transient analyses which cause a significant decrease in CPR. Types of transients evaluated were loss of flow, pressure and power increase, coolant temperature decrease, and rod withdrawal error. The most limiting transients in the stated categories are as follows:

Event	ΔCPR 104% Rated Power Exposure to 8500 MWD/T	ΔCPR 104% Rated Power 8500 MWD/T to EOC
Rod Withdrawal Error	0.16	0.16
Loss of Feedwater Heater	0.15	0.15
Turbine Trip w/o Bypass	0	0.29

With a ΔCPR of 0.16 the rod withdrawal error is the most severe abnormal operational transient during the first part of the cycle to an exposure of 8500 MWD/T. Addition of this ΔCPR to the safety limit MCPR of 1.06 gives the minimum operating limit MCPR of 1.22 required to avoid violating the safety limit, should this limiting transient occur. When an exposure of 8500 MWD/T is attained the largest ΔCPR from any transient is 0.16. When the exposure exceeds 8500 MWD/T the gradual change in scram reactivity as the rods out condition is approached results in an increasing ΔCPR until at EOC conditions the maximum ΔCPR is 0.29, caused by the turbine trip without bypass transient. The addition of the turbine trip without bypass ΔCPR to the safety limit MCPR of 1.06 gives the operating limit MCPR of 1.35 during the exposure period from 8500 MWD/T to EOC conditions.

Minimum critical power ratio operating limits are summarized below:

Cycle Exposure	MCPR Operating Limit
8500 MWD/T into the Cycle	1.22
8500 MWD/T to EOC	1.35

The transient analyses were evaluated with the scram reactivity insertion rates at both 8500 MWD/T and at EOC which include an acceptable design conservatism factor. The initial parameters used for the worst operational transient analyses were acceptable and included a CPR equal to or greater than the established operating MCPR values of 1.22 or 1.35, depending on core exposure.

Conservatism was applied in the determination of the required operating limit MCPR because the axial and local peaking were assumed to take place at the beginning of the fuel cycle and the peak of the axial power shape was assumed to occur in the mid-plane (node 12; APF of 1.40). This is the worst consistent set of parameters that is supported by a GE study (4) which has shown the required operating MCPR to be a function of the location of axial peak. The required MCPR's are essentially independent of peak location for power distributions that peak in the middle and upper portions of the core. However, for power distributions that peak near the bottom of the core, the required MCPR is reduced.

The applied R factors of 1.098 (1.154 for low enrichment bundles) for 7 x 7 fuel are taken at the beginning of cycle to reasonably bound the expected operating conditions. During the cycle the local peaking and therefore the R factors are reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced and end-of-cycle R factors, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

The ECCS performance analysis assumed that reactor operation will be limited to operating MCPR no lower than 1.18. The proposed operating limits of 1.22 or 1.35 are higher than 1.18 thus are acceptable in terms of ECCS performance.

Rod Withdrawal Error Transient

The licensees discussed the rod withdrawal error transient in terms of worst case conditions. The analysis shows that the local power range monitor subsystem (LPRM's) will detect high local powers and

alarm. However, if the operator ignores the LPRM alarm, the rod block monitor subsystem (RBM) will stop rod withdrawal while the critical power ratio is still greater than the 1.06 MCPR safety limit, and the cladding plastic strain limit of one percent is not exceeded. We conclude that the consequences of this localized transient are acceptable.

Operating MCPR Limits for Less than Rated Power and Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow condition, the licensees will conform to Technical Specifications limiting conditions for operation, Figure 3.1.1. This requires the licensees to maintain the required operating MCPR greater than 1.22 for Cycle 1 exposures up to 8500 MWD/T and greater than 1.35 from 8500 MWD/T to EOC conditions times the K_f factor for core flows less than rated. The K_f factor curves were generically derived to assure that the most limiting transient occurring at less than rated flow will not result in a MCPR below the safety limit of 1.06. We conclude that the submitted safety analyses of abnormal operational transients for FritzPatrick Nuclear Power Plant are acceptable. The minimum operating limit MCPR established for FitzPatrick that is required to avoid violation of the Safety Limit MCPR, should the most limiting transient occur, is acceptable.

Overpressure Protection

The licensees submitted an overpressure analysis in order to demonstrate that an adequate margin exists below the ASME code allowable pressure of 110% of vessel design pressure. The transient was the closure of all main steam isolation valves with high neutron flux scram. The analysis was performed based on a 104% steady state power level with the end of cycle scram reactivity applicable to the initial (current) fuel cycle, no credit for relief valve operation, and all safety valves operable. The peak pressure at the bottom of the vessel was calculated to be 1270 psig yielding a margin of 105 psig below the allowable 1375 psig ASME code limit (110% of the 1250 psig design pressure).

We find the overpressure analysis acceptable on the basis that the sensitivity study with one failed valve shows considerable margin below the allowable limit.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant evnironmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Comparison of Present Limit to Previous Limit

The controlling ΔCPR at EOC in the above evaluation is 0.29 caused by a turbine trip without bypass transient. In Reference 1 this same transient at EOC conditions yielded a ΔCPR of 0.31, as discussed in the Background section of this Safety Evaluation. The reduction from 0.31 to 0.29 is due to data from full scale tests which demonstrated that the total bypass flow is greater than was assumed for the previous analysis. The new data resulted in reduced bypass voiding, which in turn results in a change in the EOC scram reactivity curve. This change in scram reactivity resulted in the ΔCPR decrease.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

REFERENCES

1. Safety Evaluation by the Office of Nuclear Reactor Regulation for Amendment No. 14 to Facility Operating License No. DPR-59, March 12, 1976.

2. James A. FitzPatrick Nuclear Power Plant Channel Inspection and Safety Analysis with Bypass Holes Plugged, NEDO-21166, January, 1976.

3. James A. FitzPatrick Nuclear Power Plant Channel Inspection and Safety Analysis with Bypass Holes Plugged, NEDO-21166-1, April, 1976, Supplement 1.

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

Dated:

May 21, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 18 to Facility Operating License No. DPR-59 issued to the Power Authority of the State of New York and the Niagara Mohawk Power Corporation which revised the Technical Specifications for operation of the James A. FitzPatrick Nulcear Power Plant, located in Oswego County, New York. The amendment is effective as of its date of issuance.

The amendment changes the Technical Specifications to provide for a reduction in the operating minimum critical power ratio consistent with low core exposure while preserving the existing safety margin.

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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) application for amendment submitted by letter dated May 5, 1976, (2) Amendment No. 18 to License No. DPR-59, 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. and at the Oswego City Library, 120 East Second Street, Oswego, New York.

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 Operating Reactors.

Dated at Bethesda, Maryland, this 21st day of May, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors