March 29, 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. 74121)

The Commission has issued the enclosed Amendment No. 155 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 July 24, 1989.

The amendment removes requirements for the Rod Sequence Control System from the Technical Specifications and modifies the specifications associated with the Rod Worth Minimizer.

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. 155 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

PDI-1	PDI-1	OGC	PDI-1
CVogan	DLaBarge:rsc	Bachmann	RACapra
3/16/90	3/29/90	3/23/90	3/29/90

w/comments SE

DOCUMENT NAME: ISSUANCE OF AMENDMENT 74121

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

March 29, 1990

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Executive Vice President - Nuclear Generation
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123 Main Street
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for David E. LaBarge, Project Manager
Project Directorate I-1
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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. 155
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 July 24, 1989, 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.155 , 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

Acting for Daniel G. McDonald Jr.

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: March 29, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 155

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
88	88
89	89
89a	-
90	90
91	91
92	92
93	93
93a	-
94	94
95	95
96	96
97	97
98	98
99	99
99a	-
100	100
101	101
102	102
103	103
104	104
187	187

cc:

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3.3 LIMITING CONDITION FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability:

Applies to the operational status of the Control Rod System.

Objective:

To assure the ability of the Control Rod System to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive conditions during the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted.

4.3 SURVEILLANCE REQUIREMENT

4.3 REACTIVITY CONTROL

Applicability:

Applies to the surveillance requirements of the Control Rod System.

Objective:

To verify the ability of the Control Rod System to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38 percent $\Delta k/k$ the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest operable control rod fully withdrawn and all other operable rods fully inserted.

3.3.A (cont'd)

2. Reactivity margin - inoperable control rods

a. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to the Cold Shutdown condition within 24 hours and shall not be restarted unless (1) investigation has shown that the cause of the failure is not a failed control rod drive mechanism collet housing, and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.

If investigation shows that the cause of control rod failure is a cracked collet housing, or if this possibility cannot be ruled out, the reactor shall not be restarted until the affected control rod drive has been replaced or repaired.

4.3.A (cont'd)

2. Reactivity margin - inoperable control rods

a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 30 percent power. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above 30 percent power.

b. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days (these valves may be closed intermittently for testing under administrative control).

c. The status of the pressure and level alarms for each accumulator shall be checked once per week.

d. When it is initially determined that a control rod is incapable of normal insertion, an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted, shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted. If Specification 3.3.A.1 and 4.3.A.1 are met, reactor startup may proceed.

3.3.A.2 (cont'd)

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

d. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

e. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reactor power operation, no more than one control rod in any 5 X 5 array may be inoperable (at least 4 operable control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a cold condition within 24 hours.

4.3.A.2 (cont'd)

e. The scram discharge volume drain and vent valves shall be full-travel cycled at least once per quarter to verify that the valves close in less than 30 seconds and to assure proper valve stroke and operation.

3.3 (cont'd)

B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

4.3 (cont'd)

B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

   a. When a rod is withdrawn the first time after each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is above 20 percent power shall be performed to verify instrumentation response.

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

   c. During each refueling outage and after each control rod maintenance, observe that the drive does not go to the overtravel position.

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

3.3.B (cont'd)

3. 'Whenever the reactor is below 10% rated thermal power, ·e Rod Worth Minimizer (RWM) shall be operable except ·s follows:

   a. Should the RWM become inoperable during a reactor startup after the first twelve control rods have been withdrawn, or during a reactor shutdown, control rod movement may continue provided that a second licensed reactor operator, licensed senior operator, or reactor engineer independently verifies that the control rods are being positioned in accordance with the RWM program sequence.

   b. Should the RWM be inoperable before a startup is begun, or become inoperable during the withdrawal of the first twelve control rods, the startup may continue provided that a reactor engineer independently verifies that the control rods are being positioned in accordance with the RWM program sequence. After twelve control rods have been fully withdrawn, startup may continue in accordance with Specification 3.3.B.3.a above.

4.3.B (cont'd)

3. The capability of the Rod Worth Minimizer to properly fulfill its function shall be demonstrated by the following checks:

   a. During startup, prior to the start of control rod withdrawal:

      (1) The correctness of the RWM program sequence shall be verified.

      (2) The RWM computer on line diagnostic test shall be successfully performed.

      (3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be demonstrated.

      (4) The rod block function of the RWM shall be demonstrated by withdrawing an out-of-sequence control rod no more than to the block point, then reinserting the subject rod.

   b. During shutdown, prior to attaining 10% rated power during rod insertion, except by scram:

      (1) The correctness of the RWM program sequence shall be verified.

      (2) The RWM computer on line diagnostic test shall be successfully performed.

3.3.B.3 (cont'd)

4.3.B (cont'd)

c. When required by Specifications 3.3.B.3.a or b, the second licensed reactor operator, licensed senior operator, or the reactor engineer must be present at the reactor console during rod movements to verify compliance with the prescribed rod pattern. This individual shall have no other concurrent duties during the rod withdrawal or insertion.

d. Plant startup under Specification 3.3.B.3.b is only permitted once per calendar year. Any startup conducted without the RWM as described in Specification 3.3.B.3.b shall be reported to the NRC within 30 days of the startup. This special report shall state the reason for the RWM inoperability, the action taken to restore it, and the schedule for returning the RWM to an operable status.

e. Control rod patterns shall be equivalent to those prescribed by the Banked Position Withdrawal Sequence (BPWS) such that the drop of any in-sequence control rod would not result in a peak fuel enthalpy greater than 280 calories/gm.

f. If Specifications 3.3.B.3.a through e cannot be met, the reactor shall not be restarted, or if the reactor is in the run or startup modes at less than 10% rated thermal power, no rod movement is permitted except by scram.

3.3.B (cont'd)

4. Control rods shall not be withdrawn for startup or during refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second except as permitted by Specifications 3.10.B.3 and 3.10.B.4.

5. During operation with limiting control rod patterns, as determined by the reactor engineer, either:

   a. Both RBM channels shall be operable, or

   b. Control rod withdrawal shall be blocked, or

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

4.3.B (cont'd)

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second except as permitted by Specifications 3.10.B.3 and 3.10.B.4.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).

3.3 (cont'd)

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

Control Rod Notch Position Observed	Average Scram Insertion Time (Seconds)
46	0.338
38	0.923
24	1.992
04	3.554

4.3 (cont'd)

C. Scram Insertion Times

1. After each refueling outage, all operable rods shall be scram time tested from the fully withdrawn position with the nuclear system pressure above 950 psig (with saturation temperature). This testing shall be completed prior to exceeding 40% power. During all scram time testing below 10% power, the RWM shall be operable.

3.3.C (cont'd)

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Control Rod Notch Position Observed	Average Scram Insertion Time (Seconds)
46	0.361
38	0.977
24	2.112
04	3.764

3. The maximum scram insertion time for 90 percent insertion of any operable control rod shall not exceed 7.00 sec.

4.3.C (cont'd)

2. At 16-week intervals, 10 percent of the operable control rod drives shall be scram timed above 950 psig. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. All control rods shall be determined operable once each operating cycle by demonstrating the scam discharge volume drain and vent valves operable when the scram test initiated by placing the mode switch in the SHUTDOWN position is performed as required by Table 4.1-1 and by verifying that the drain and vent valves:

   a. Close in less that 30 seconds after receipt of a signal for control rods to scram, and

   b. Open when the scram signal is reset or the scram discharge instrument volume trip is bypassed.

3.3 (cont'd)

D. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1 percent Δ k. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

E. If Specifications 3.3.C and D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold condition within 24 hr.

4.3 (cont'd)

D. Reactivity Anomalies

During the Startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

3.3 and 4.3 BASES

A. Reactivity Limitation

1. The requirements for the Control Rod Drive System have been identified by evaluating the need for reactivity control via control rod movement over the full spectrum of plant conditions and events. As discussed in subsection 3.6 of the FSAR, the Control Rod System design is intended to provide sufficient control of core reactivity so that the core could be made subcritical with the strongest rod fully withdrawn. This reactivity characteristic has been a basic assumption in the analysis of plant performance. Compliance with this requirement can be demonstrated conveniently only at the time of initial fuel loading or refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration shall be performed with the reactor core in the cold, xenon-free condition and will show that the reactor is subcritical by at least R + 0.38 %Δ k/k with the analytically determined strongest control rod fully withdrawn.

The value of "R", in units of % Δ k/k, is the amount by which the core reactivity, in the most reactive condition at any time in the subsequent operating cycle, is calculated to be greater than at the time of the demonstration. "R", therefore, is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of "R" must be positive or zero and must be determined for each fuel cycle.

The demonstration is performed with a control rod which is calculated to be the strongest rod. In determining this analytically strongest rod, it is assumed that every fuel assembly of the same type has identical material properties. In the actual core, however, the control cell material properties vary within allowed manufacturing tolerances, and the strongest rod is determined by a combination of

3.3 and 4.3 BASES (cont'd)

the control cell geometry and local k_{∞} . Therefore, an additional margin is included in the shutdown margin test to account for the fact that the rod used for the demonstration (the analytically strongest) is not necessarily the strongest rod in the core. Studies have been made which compare experimental criticals with calculated criticals. These studies have shown that actual criticals can be predicted within a given tolerance band. For gadolinia cores the additional margin required due to control cell material manufacturing tolerances and calculational uncertainties has experimentally been determined to be 0.38% Δk. When this additional margin is demonstrated, it assures that the reactivity control requirement is met.

2. Reactivity Margin - Inoperable Control Rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted, it is in a safe position of maximum contribution to shutdown reactivity. If it is in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shut down at all times with the remaining control rods assuming the strongest operable control rod does not insert.

Inoperable bypassed rods will be limited within any group to not more than one control rod of a (5x5) twenty-five control rod array.

Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing will assure that the reactor will not be operated with a large number of rods with failed collet housings.

B. Control Rods

1. Control rod drop accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. The overtravel

3.3 and 4.3 BASES (cont'd)

position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement could indicate an uncoupled condition. Rod position indication is required for proper function of the Rod Worth Minimizer (RWM).

2. The control rod housing support restricts the outward movement of a control rod to less than 3 in. in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the Primary Coolant System. The design basis is given in subsection 3.8.2 of the FSAR, and the safety evaluation is given in subsection 3.8.4. This support is not required if the Reactor Coolant System is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated, since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. The Rod Worth Minimizer (RWM) restricts the order of control rod withdrawal and insertion to be equivalent to the Banked Position Withdrawal Sequence (BPWS). These sequences are established such that the drop of any in-sequence control rod from the fully inserted position to the position of the control rod drive would not cause the reactor to sustain a power excursion resulting in a peak fuel enthalpy in excess of 280 cal/gm. An enthalpy of 280 cal/gm is well below the level at which rapid fuel dispersal could occur (i.e. 425 cal/gm.). Primary system damage in this accident is not possible unless a significant amount of fuel is rapidly dispersed. Ref. Subsections 3.6.6, 7.7.4.3 and 14.6.1.2 of the FSAR, NEDE-24011 and NEDO-10527 including supplements 1 and 2 to NEDO-10527.

In performing the function described above, the RWM is not required to impose any restrictions at core power levels in excess of 10% of rated. Material in the cited references shows that it is impossible to reach 280 calories per gram in the event of a control rod drop occurring at power greater than 10%, regardless of the rod pattern. This is true for all normal and abnormal patterns including those which maximize the individual control rod worth.

3.3 and 4.3 BASES (cont'd)

At power levels below 10% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calories per gram drop limit. In this range, the RWM constrains the control rod sequence and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviance from planned withdrawal sequences. It serves as a backup to procedural control of control rod sequences which limit the maximum reactivity of control rods. Normal RWM program aborts do not constitute an inoperable condition if the RWM can be reinitialized. In the event that the Rod Worth Minimizer is out of service, a second licensed reactor operator, licensed senior operator or reactor engineer can manually fulfill the control rod pattern conformance functions of this system.

Below 10% of rated power, the RWM forces adherence to acceptable rod patterns. Above 10% of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 10% of rated power are imposed by power distribution requirements as specified in Sections 3.1.B, 3.5.H, and 3.5.I of these Technical Specifications.

Power level for automatic cutout of the RWM function is sensed by steam flow and is manually set above 10% of rated power to account for instrument error.

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 10% rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

4. The Source Range Monitor (SRM) System performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per sec. assures that 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.

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 (i.e., MCPR limits as shown in specification 3.1.B). 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.

3.3 and 4.3 BASES (cont'd)

In the analytical treatment of the transients, 290 msec are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 msec estimated from the scram test results. Approximately 90 msec of each of these intervals result from the sensor and the circuit delay. At this point, the scram pilot valve solenoid de-energizes. Approximately 120 msec later, control rod motion is estimated to actually begin. However, 200 msec is conservatively assumed for this time interval in the transient analysis and this is also included in the allowable scram insertion times of Specification 3.3.C. The time to de-energize the scram pilot valve solenoid is measured during the calibration tests required by Specification 4.1.

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 instance 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 the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% Δk. Deviations in core reactivity greater than 1% Δk are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

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3.7 BASES

A. Primary Containment

The integrity of the primary containment and operation of the Emergency Core Cooling Systems in combination limit the offsite doses to values less than those specified in 10 CFR 100 in the event of a break in the Reactor Coolant System piping. Thus, containment integrity is required whenever the potential for violation of the Reactor Coolant System integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception to the requirement to maintain primary containment integrity is allowed during core loading and during low power physics testing when ready access to the reactor vessel is required. There will be no pressure on the system at this time, which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period, however, restrictive operating procedures and operation of the RWM in accordance with Specification 3.3.B.3 minimize the probability of an accident occurring. Procedures in conjunction with the Rod Worth Minimizer Technical Specifications limit individual control worth such that the drop of any in-sequence control rod would not result in a peak fuel enthalpy greater than 280 calories/gm. In the unlikely event that an excursion did occur, the reactor building and Standby Gas Treatment System, which shall be operational during this time, offers a sufficient barrier to keep offsite doses well within 10 CFR 100.

The pressure suppression pool water provides the heat sink for the Reactor Coolant System energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1,020 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 56 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Section 5.2).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 155 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 July 24, 1989 (Reference 1), the Power Authority of the State of New York (the licensee), requested an amendment to Facility Operating License No. DRP-59 for the James A. FitzPatrick Nuclear Power Plant. The proposed amendment would change Technical Specification (TS) 3/4.3 and associated Bases to permit removal of the Rod Sequence Control System (RSCS), reduce the Rod Worth Minimizer (RWM) low power setpoint, enhance operation of the RWM, implement Banked Position Withdrawal Sequence (BPWS) rod withdrawal, and provide several minor administrative changes.

DISCUSSION

The Rod Sequence Control System is designed to restrict rod movement to minimize the individual worth of control rods to lessen the consequences of a Rod Drop Accident (RDA). Control rod movement is restricted through the use of rod select, insert, and withdraw blocks. The RSCS is a hardwired (as opposed to a computer controlled), redundant backup to the RWM. It is independent of the RWM in terms of inputs and outputs but the two systems are compatible. The RSCS is designed to monitor and, when necessary, block operator control rod selection, withdrawal and insertion actions. This blocking action is designed to assist in preventing significant control rod pattern errors which could result in a control rod with a high reactivity worth that might cause fuel damage if dropped. A significant pattern error is one of several abnormal events all of which must occur to have an RDA which might exceed fuel energy density limit criteria for the event. It was designed only for possible mitigation of the RDA and is active only during low power operation (currently generally less than 20 percent power) when an RDA might be significant. It provides rod blocks on detection of a significant pattern error, but does not prevent an RDA. A similar pattern control function is also performed by the RWM, a computer controlled system. All reactors having an RSCS also have an RWM.

In August 1986 (Reference 2), the BWR Owners' Group (BWROG) in cooperation with the General Electric Company, proposed Amendment 17 to GESTAR II (Reference 3) which would eliminate the requirement for the RSCS and retain the RWM, but lower the setpoint for turnoff (during startup) or turnon (during

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shutdown) from 20 to 10 percent. The NRC staff review concluded that the proposed changes were acceptable, and approved Amendment 17, but imposed several additional requirements which would be necessary to implement the changes. The staff safety analysis and additional requirements were presented and discussed in an attachment to Reference 4. (This review and approval is also available in Reference 3, page US.C-379.)
The additional requirements were:

1) The Technical Specifications should require provisions for minimizing operations without the RWM system operable.

2) The occasional necessary use of a second operator replacement should be strengthened by a utility review of relevant procedures, related forms and quality control to assure that the second operator provides an effective and truly independent monitoring process. A discussion of this review should accompany the request for RSCS removal.

3) Rod patterns used should be at least equivalent to Banked Position Withdrawal Sequence (BPWS) patterns.

EVALUATION

The licensee has proposed changes to TS 3/4.3, and associated Bases, in four categories to accomplish the changes and to meet the requirements discussed above. These changes are:

A. Elimination of the RSCS requirements.

B. Reduction of the RWM setpoint to 10 percent.

C. Increased administrative control of RWM operability (intended to result in decreased use of the second operator as a substitute for the RWM), and implementation of BPWS. The licensee has also discussed the procedures for second operator actions when required, to ensure independent monitoring of the control rod patterns.

D. Administrative changes, correcting errors, relocating text and improving the clarity of the text.

The NRC staff review and basis for approval of the removal of the RSCS and lowering of the setpoint for the RWM, as proposed by the licensee in sections of the submittal relating to topics A and B above, are provided in References 2 or 3. The proposed changes fall within the scope of that staff review and approval. The present staff review of the proposed TS changes that implement these operational changes concludes that they are appropriate, clearly stated and are acceptable.

The licensee has increased the administrative control of the RWM, as required in the staff review of RSCS removal. The proposed revised TS require the RWM to be operable at the beginning of each startup, with only one exception per year. This follows the pattern of previously approved RWM TS for BWR 3 operation (discussed in Reference 4). These have been found to provide the desired improvement in reliability for the system. A report to the NRC is required by the TS whenever the RWM is inoperable for startup. This will indicate corrective actions to improve reliability. Also, as required, the TS and procedures for the use of a second operator (when the RWM is inoperable) have been reviewed and improved where necessary and have been discussed in the submittal, and appear to provide a suitable independent check on the rod patterns. Finally, as required, the revised TS prescribe the use of rod patterns equivalent to the BPWS patterns approved by previous staff reviews to maintain low control rod reactivity worths. The changes and reviews are in accord with the staff requirements of Reference 4 and are acceptable, and the proposed changes to TS 3/4.3 and Bases appropriately implement the changes.

The administrative changes relating to topic D above are applicable to TS 3/4.3 and Bases, are primarily correction of minor errors and improvements in clarity or format. The changes include the relocation of Specification 3.3.A.2.c to 4.3.A.2.d and the correction of an error in the Bases on the parameter sensed for the RWM setpoint, both of which have been previously submitted for review, and the change to the Bases 3.7.A which reflects better correspondence to existing TS 3.7.A.2 and to the changes to TS 3/4.3. The review has indicated that the proposed changes are appropriate and acceptable.

SUMMARY

The staff has reviewed the amendment submitted by the licensee for the FitzPatrick plant proposing TS changes relating to the removal of the RSCS. Based on this review, we have concluded that appropriate documentation was submitted and the proposed TS changes satisfy staff positions and requirements in these areas. Operation in the modes proposed for FitzPatrick is acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment changes 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 surveillance 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. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). 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.

REFERENCES

1. Letter and Enclosures from J. Brons, New York Power Authority, to USNRC, dated July 24, 1989, "Proposed changes to the Technical Specifications Regarding Elimination of the Rod Sequence Control System."

2. Letter and Enclosures from T. A. Pickens, BWR Owners' Group to G. Lainas, NRC, dated August 15, 1986, "Amendment 17 to GE Licensing Topical Report NEDE-24011-P-A."

3. NEDE-24011-P-A-9, September 1988, "General Electric Standard Application for Reactor Fuel," (GESTAR II).

4. Letter from A. Thadani, NRC, to J. Charnley, General Electric, dated December 27, 1987, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, Revision 8, Amendment 17."

operation (discussed in Reference 4). These have been found to provide the desired improvement in reliability for the system. A report to the NRC is required by the TS whenever the RWM is inoperable for startup. This will indicate corrective actions to improve reliability. Also, as required, the TS

Dated: March 29, 1990

PRINCIPAL CONTRIBUTORS:
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