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. 53
License No. DPR-59

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

   A. The applications for amendment by Power Authority of the State of New York (the licensee) dated January 6, 1981 and February 20, 1981 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commissions'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 Commissions'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 Commissions'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 3.B of Facility Operating License No. DPR-59 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 53, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 13, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 53

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
32	32
56	56
76a	76a
79	79
101	101
102	102
159	159

3.1 BASES

The reactor protection system automatically initiates a reactor scram to:

1. Preserve the integrity of the fuel cladding.
2. Preserve the integrity of the Reactor Coolant System.
3. Minimize the energy which must be absorbed following a loss of coolant accident, and prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The Reactor Protection System is of the dual channel type (Reference subsection 7.2 FSAR). The System is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the intent of IEEE - 279 (1971) for Nuclear Power Plant Protection Systems. The system has a reliability greater than that of a 2 out of 3 system and somewhat less than that of a 1 out of 2 system.

With the exception of the average power range monitor (APRM) channels, the intermediate range monitor (IRM) channels, the main steam isolation valve closure and the turbine stop valve closure, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved.

Three APRM instrument channels are provided for each protection trip system. APRM's A and E operate contacts in one subchannel and APRM's C and E operate contacts in the other steam line isolation valves, main steam drain valves, recirc. sample valves (Group 1), initiates the HPCI and RCIC and trips the recirculation pumps. The low-low-low reactor water level instrumentation is set to trip when the water level is 18 in. above the top of the active fuel. This trip activates the remainder of the ECCS subsystems, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate ECCS operation and primary system isolation so that post-accident cooling can be accomplished and the guidelines of 10CFR100 will not be exceeded. For large breaks up to the complete circumferential break of a 24 in. recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference paragraph 6.5.3.1 FSAR.

The high drywell pressure instrumentation is a diverse signal for malfunctions to the water level instrumentation and in addition to initiating ECCS, it causes isolation of Groups B and 3 isolation valves. For the breaks discussed above, this instrumentation will generally initiate ECCS operation before the low-low-low water level instrumentation; thus the results given above are applicable here also. See Specification 3.7 for isolation valve closure group. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperature peak at approximately 1,000°F and release of radioactivity to the environs is below 10CFR100 guidelines. Reference Section 14.6.5 FSAR.

TABLE 3.2-6

SURVEILLANCE INSTRUMENTATION

Minimum No. of Operable Instrument Channels	Instrument	Type Indication and Range	No. of Channels Provided by Design	Action
1	(Suppression Chamber (Water Level ((Wide Range) ((Suppression Chamber (Water Level ((Narrow Range)	Indicator) Recorder) -72 to +72 inches)) Indicator) Recorder) -6 to +6 inches)	2	(2)
N/A	Control Rod Position Indication	Indicator Postion 00 to 48	1	(7)
2	Source Range Monitors	Indicator Recorder 1 to 10^6 cps	4	(8)
3	Intermediate Range Monitor	Indicator Recorder 10^{-4} to 40% Rated Power	8	(8)(9)
2	Average Power Range Monitor	Indicator Recorder 0-125% Rated Power	6	(8)(9)
1	Drywell-Suppression Chamber Differential Pressure	Recorder 0 to 5 psi Computer 0 to 5 psi	2	(2)

NOTES FOR TABLE 3.2-6

1. From and after the date that the minimum number of operable instrument channels is one less than the minimum number specified for each parameter, continued operation is permissible during the succeeding 30 days unless the minimum number specified is made operable sooner.

2. In the event that all indications of this parameter is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

TABLE 4.2-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor Water Level	(1)	Once/3 months	Once/day
2) Drywell Pressure	(1)	Once/3 months	None
3) Reactor Pressure	(1)	Once/3 months	None
4) Auto Sequencing Timers	NA	Once/operating cycle	None
5) ADS - LPCI or CS Pump Disch. Pressure Interlock	(1)	Once/3 months	None
6) Trip System Bus Power Monitors	(1)	N/A	None
8) Core Spray Sparger d/p	(1)	Once/3 months	Once/day
9) Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
10) Steam Line/Area High Temp.(HPCI & RCIC)	(1)	Once/operating cycle	Once/day
12) HPCI & RCIC Steam Line Low Pressure	(1)	Once/3 months	None
13) HPCI Suction Source Levels	(1)	Once/3 months	None
14) 4KV Emergency Power Under-Voltage Relays and timers	Once/operating cycle	Once/operating cycle	None
15) HPCI & RCIC Exhaust Diaphragm Pressure High	(1)	Once/3 months	None
17) LPCI/Cross Connect Valve Position	Once/operating cycle	NA	NA

Note: See listing of notes following Table 4.2-6 for the notes referred to herein.

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

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviance from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences which limit the maximal reactivity worth of control rods , in the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator or other qualified technical plant employee

can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural control to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20%, these devices force adherence to acceptable rod patterns. Above 20% of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20% of rated power are imposed by power distribution requirements as defined in Section 3.3.3.5 of these Technical Specifications. Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of ± 2% of full power, the nominal instrument setting is 22% of rated power. Power level for automatic cutout of the RWM function is sensed by feedwater and steam flow and is set manually at 30% of rated power to be consistent with the RSCS setting.

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

The RSCS can be functionally tested prior to control rod withdrawal for reactor startup. By selecting, for example, A_{12} and attempting to withdraw, by one notch, a rod or all rods in each other group, it can be determined that the A_{12} group is exclusive. By bypassing to full-out all A_{12} rods, selecting A_{34} and attempting to withdraw, by one notch, a rod or all rods in group B, the A_{34} group is determined exclusive. The same procedure can be repeated for the B groups. After 50% of the control

3.3 and 4.3 BASES (cont'd)

rods have been withdrawn (e.g., groups A_{12} and A_{34}), it is demonstrated that the Group Notch mode for the control drives is enforced. This demonstration is made by performing the hardware functional test sequence. The Group Notch restraints are automatically removed above 20% power.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCS occurs and subsequently at appropriate stages of the control rod insertion.

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.

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 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 qualified personnel may perform this function.

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 the Safety Limit. Scram insertion time and scram reactivity curves shown in NEDO-24242, Figures 2a, 2b and 2c were used in analyses of power transients to determine MCPR limits. The scram insertion time test criteria of Section 3.3.C.1 conform to the scram insertion times of NEDO-24242. Therefore, the required protection is provided.

Item No.	Examination Category Table IS-251	Components and Parts to be Examined	Examination Method	Extent of Examination % in 10 Years	Extent of Examination, % Intervals 10 Years	5 Years	Accessibility	Comments and Examination Methods
1.6	E-2 Pressure-containing welds in vessel penetrations	Welds in vessel at control rod drive penetrations and in-core monitor housing (Stud tube-to-housing and vessel)	Visual	25	25	0	Access is provided by observation ports in bottom head insulation.	Visual examination will be performed using optical equipment capable of providing a complete viewing of the O.D. of the housing external to the vessel for signs of leakage.
1.7	F Pressure-containing dissimilar metal welds.	Primary nozzles to safe-end welds: Recirculation inlet Recirculation outlet Closure nozzles Core spray	Visual, surface, and volumetric	100	100	33	Reactor pressure vessel safe-end welds are made accessible by removing thermal insulation and sacrificial shield plugs.	Remote or local visual examination will be performed. The extent to which surface examination is performed is determined by radiological considerations. Manual ultrasonic examination will be performed, where possible, until automated equipment is available but the extent of the examination will be determined by radiological considerations.
4.1	Pressure-containing dissimilar metal welds.	Piping pressure boundary safe-ends in branch piping welds Piping pressure boundary welds between dissimilar metals.	Visual, surface, and volumetric	100	100	33	Safe ends in branch welds and dissimilar metals welds are made accessible by removing piping thermal insulation.	Remote or local visual examination will be performed. The extent to which surface examination is performed is determined by radiological considerations. Although radiation dosage is high, manual ultrasonic examination will be performed, where possible, until automated equipment becomes available, but the extent of the examination will be determined by radiological considerations. Exception is taken to volumetric examination of welds which require drainage of the reactor vessel.
5.3	Pressure-containing dissimilar metal welds	Pump pressure boundary nozzles-to-safe end welds	Visual and volumetric	100	0	0	Not applicable.	Not applicable. There are no nozzles to Safe-end welds on pumps.
6.3	Pressure-containing dissimilar metal welds	Valve pressure boundary valve to safe-end welds	Visual and volumetric	100	100	33	Safe ends in valves are made accessible by removing thermal insulation.	Remote or local visual examination will be performed. Although radiation dosage is high, manual ultrasonic examination will be performed, where possible, until automatic equipment becomes available. Exception is taken to volumetric examination of welds which require drainage of reactor vessel.