ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

RELATED TO

NUREG-0578 TMI-2 LESSONS LEARNED

CATEGORY "A" ITEMS

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333
October 10 , 1980

8010160500

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 Position 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 Monitors	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)
1	Safety/Relief Valve Position Indicator (10)	Indicator Open/Closed	2	(2) (11)

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.

NOTES FOR TABLE 3.2-6 (CONTINUED)

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.

3. Three (3) indicators from level instrument channel A, B, & C. Channel A or B are utilized for feedwater control, reactor water high and low level alarms, recirculation pump runback. High level trip of main turbine and feedwater pump turbine utilizes channels A, B, & C.

4. One (1) recorder utilized the same level instrument channel as selected for feedwater control.

5. Three (3) indicators from reactor pressure instrument channel A, B, & C. Channel A or B are utilized for feedwater control and reactor pressure high alarm.

6. One (1) recorder. Utilizes the same reactor pressure instrument channel as selected for feedwater control.

7. The position of each of the 137 control rods is monitored by the Rod Position Information System. For control rods in which the position is unknown, refer to Paragraph 3.3.A.

8. Neutron monitoring operability requirements are specified by Table 3.1-1 and Paragraph 3.3.B.4.

9. A minimum of 3 IRM or 2 APRM channels respectively must be operable (or tripped) in each safety system.

10. Each Safety Relief Valve position indicator is equipped with two acoustical detectors of which one is in service and a backup thermocouple detector.

11. From and after the date that none of the acoustical detectors is operable but the thermocouple detector is operable, continued operation is permissible during the succeeding 30 days unless one acoustical detector is made operable sooner.

TABLE 4.2-6

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLIANCE INSTRUMENTATION

	INSTRUMENT CHANNEL	CALIBRATION FREQUENCY	INSTRUMENT CHECK
1.)	Reactor Water Level	Once/6 months	Once Each Shift
2.)	Reactor Pressure	Once/6 months	Once Each Shift
3.)	Drywell Pressure	Once/6 months	Once Each Shift
4.)	Drywell Temperature	Once/6 months	Once Each Shift
5.)	Suppression Chamber Temperature	Once/6 months	Once Each Shift
6.)	Suppression Chamber Water Level	Once/6 months	Once Each Shift
7.)	Control Rod Position Indication	N/A	Once Each Shift
8.)	Neutron Monitoring (APRM)	Five/week	Once Each Shift
9.)	Neutron Monitoring (IRM and SRM)	Note 10	Note 10
10.)	Drywell-Suppression Chamber Differential Pressure	Once/6 months	Once Each Shift
11.)	Safety/Relief Valve Position Indicator (Primary)	Note 11	Once Each Week

NOTES FOR TABLES 4.2-1 THROUGH 4.2-6

1. Initially once every month until acceptance failure rate data are available; thereafter, a request may be made to the NRC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operate in a environment similar to that of JAFNPP.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed prior to each startup or prior to preplanned shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

   These instrument channels will be calibrated using simulated electrical signals once every three months.

4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

5. Reactor low water level, high drywell pressure and high radiation main steam line tunnel are not included on Table 4.2-1 since they are tested on Table 4.1-2.

6. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

7. At least one (1) Main Stack Dilution Fan is required to be in operation in order to isokinetically sample the Main Stack.

8. Uses same instrumentation as Main Steam Line High Radiation. See Table 4.1-2.

9. See Technical Specification 1.0.F.4, Definitions, for meaning of term, "Instrument Check".

10. Calibration and instrument check surveillance for SRM and IRM Instruments are as specified in Table 4.1-1, 4.1-2, 4.2-3.

11. Functional test is performed once each operating cycle.

6. In addition to items 1, 2 & 3 above, two additional operators shall be readily available on site whenever the reactor is in other than cold shutdown. During cold shutdown, an additional operator shall be readily available on site.

7. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

8. In the event of illness or absenteeism up to two (2) hours is allowed to restore the shift crew or fire-brigade to normal complement.

9. A Fire Brigade of five (5) or more members shall be maintained on site at all times. This excludes two (2) members of the minimum shift crew necessary for safe shutdown and any personnel required for other essential functions during a fire emergency.

10. A Shift Technical Advisor shall be on site and readily available to the control room except during the cold shutdown or refuel mode.

6.3 PLANT STAFF QUALIFICATIONS

The minimum qualifications with regard to educational background and experience for plant staff positions shown in Fig. 6.2-1 shall meet or exceed the minimum qualifications of ANSI NI8.1-1971 for comparable positions; except for the Radiation and Environmental Services Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents. Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.

6.4 RETRAINING AND REPLACEMENT TRAINING

A training program shall be maintained under the direction of the Training Coordinator to assure overall proficiency of the plant staff organization. It shall consist of both retraining and replacement training and shall meet or exceed the minimum requirements of Section 5.5 of ANSI NI8. 1-1971.

The retraining program shall not exceed periods two years in length with a curriculum designed to meet or exceed the requalification requirements of 10 CFR 55, Appendix A. In addition fire brigade training shall meet or exceed the requirements of NFPA 27-1975, except for Fire Brigade training sessions which shall be held at least quarterly. The effective date for implementation of fire brigade training is March 17, 1978.

6.5 REVIEW AND AUDIT

Two seperate review groups for the review and audit of plant operations have been constituted. One of these, the Plant Operating Review Committee (PORC), is an onsite group. The other is an independent review and audit group, the offsite Safety Review Committee (SRC).

6.5.1 PLANT OPERATING REVIEW COMMITTEE(PORC)

(A) Membership

The PORC is comprised of the Resident Manager (Chairman), Superintendent of Power (Vice Chairman), Operations Superintendent, Maintenance Superintendent, Technical Services Superintendent, Instrument and Control Superintendent, Radiological and Environmental Services Superintendent and Reactor Analyst. Special consultant to provide expert advice may be utilized when the nature of a particular problem dictates.

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(B) Alternates

Alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in PORC activities at any one time.

(C) Meeting Frequency

Meetings will be called by the Chairman as the occasions for review or investigation arise. Meetings will be no less frequent than once a month.

(D) Quorum

The Chairman or Vice Chairman and four members, including designated alternates, shall constitute a quorum.

(E) Responsibilities

1. Review plant procedures, and changes thereto, required by Specification 6.8.

2. Review proposed tests and experiments that affect nuclear safety.

3. Review proposed changes to the Operating License and Technical Specifications.

4. Review proposed changes or modifications to plant systems or equipment that affect nuclear safety

5. Investigate violations of the Technical Specifications and prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Resident Manager, who will forward the report to the Manager - Nuclear Operations and to the Chairman of the Safety Review Committee.

6. Review plant operations to detect potential safety hazards.

7. Review the Security Plan and implementing procedures annually.

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Amendment No. 25, 34

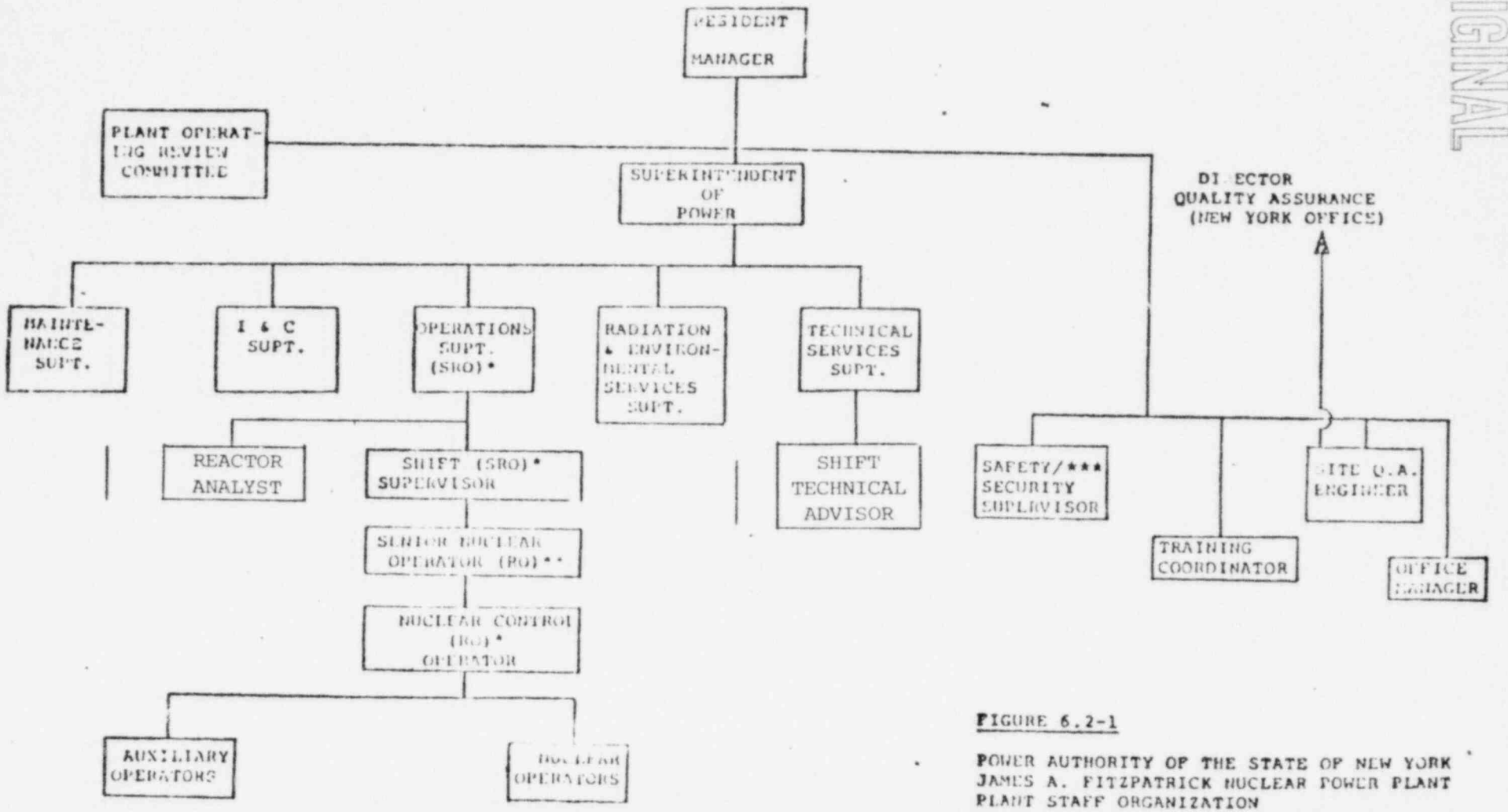

FIGURE 6.2-1

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
PLANT STAFF ORGANIZATION
*SRO - SENIOR REACTOR OPERATOR
**RO - REACTOR OPERATOR
*** Responsible for J. A. FitzPatrick Fire Protection Program

ATTACHMENT III

SAFETY EVALUATION OF

CHANGES RELATED TO NUREG-0578 TMI-2

LESSONS LEARNED CATEGORY "A" ITEMS

POWER AUTHORITY OF THE STATE OF NEW YORK
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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Section I - Description of Modification

The James A. FitzPatrick Operating License is being amended by adding license conditions related to a System Integrity Measurements Program and Improved Iodine Measurements capability. Technical Specification Sections 3.2 (Table 3.2-6) and 4.2 (Table 4.2-6) are being changed to incorporate limiting conditions for operation and surveillance requirements for the Safety Relief Valve position indicators. Technical Specification Section 6.2 and 6.3 (including Figure 6.2-1) were revised to reflect the addition of a Shift Technical Advisor to the plant staff. These changes were requested by the NRC staff in a letter dated July 2, 1980. Figure 6.2-1 has also been corrected to more accurately reflect the Reactor Analyst's position i.e., much greater involvement with operations personnel than with Technical Services personnel.

Section II - Purpose of Modification

The purpose of this modification is to incorporate the requirements of the NUREG-0578 TMI-2 Lessons Learned Category "A" items into the James A. FitzPatrick Operating License and Technical Specifications as requested in the NRC letter of July 2, 1980.

Section III - Impact of the Change

These modifications will not alter the conclusions reached in the FSAR and SER accident analysis.

Section IV - Implementation of the Modification

The modifications as proposed will not impact the ALARA or Fire Protection Program at JAF.

Section V - Conclusion

The incorporation of these modifications: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; and c) will not reduce the margin of safety as defined in the basis for any Technical Specification, and d) does not constitute an unreviewed safety question.

Section VI - References

(a) JAF FSAR

(b) JAF SER