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January 24, 19

Docket No. 50-333

Posted

Correction to Amdt. 149 to DPR-59

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

ACRS (10)
JLinville
RCapra

Dear Mr. Brons:

SUBJECT: TECHNICAL SPECIFICATION REPLACEMENT PAGES

Enclosed are selected pages to the Technical Specifications for the James A. FitzPatrick Nuclear Power Plant which contain changes which were approved and issued when the appropriate amendment was issued. However, for these pages it has been found that a later amendment which also affected the page, failed to incorporate the changes when the final amendment pages were prepared by your office and issued by the NRC. A list of the pages affected, the amendments which approved the changes, the amendments which inadvertently deleted the changes, the TAC numbers of the amendments which approved the changes, and the dates the amendments which approved the changes were issued, are shown in Enclosure 1. For the amendments listed in the "DEL. BY" Column, please replace the affected TS pages with the enclosed TS pages.

In addition, a corrected page 143 is included which corrects the spelling of "challenges" which was inadvertently introduced in the submittal for Amendment No. 134.

We regret any inconvenience this may have created.

Sincerely,

Original signed by

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

Enclosure:
As stated

cc w/enclosure:
See next page

OFC	:PDI-1	:PDI-1	:PDI-1	:	:	:	:
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DATE	:1/23/90	:1/23/90	:1/24/90	:	:	:	:

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Document Name: TECH SPECS REPLACEMENTS

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ENCLOSURE 1

LIST OF AFFECTED PAGES

PAGE	AMENDMENT NO. APP. IN	AMENDMENT NO. DEL. BY	TAC NO.	ISSUE DATE
iii	130	137	54533	5/31/89
vi	130	134	54533	5/31/89
vii	134	137	68310	7/19/89
57	134	147	68310	7/19/89
123	132	134	73827	6/22/89
125	148	149	73345	12/26/89
244a	134	135	68310	7/19/89
247	130	137	54533	5/31/89
247a	130	137	54533	5/31/89
248	134	137	68310	7/19/89

EXAMPLE: Remove page iii from Amendment No. 137 and replace it with the attached.

TABLE OF CONTENTS (Cont'd)

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LIST OF TABLES (Cont'd)

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LIST OF FIGURES

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3.2 BASES (cont'd)

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.1.2 FSAR. During the Hydrogen Addition Test, the normal background Main Steam Line Radiation Level is expected to increase by approximately a factor of 5 at the peak hydrogen concentration as indicated in note 16, Table 3.1-1. With the hydrogen addition, the fission product release would still be well within the 10 CFR 100 guidelines in the event of a control rod drop accident.

Pressure instrumentation is provided to close the main steam isolation valves in the run mode when the main steam line pressure drops below 825 psig. The reactor pressure vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the run mode is less severe than the loss of feedwater analyzed in Section 14.5 of the FSAR, therefore, closure of the main steam isolation valves for thermal transient protection when not in the run mode is not required.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

The trip settings of approximately 300 percent of design flow for this high flow or 40°F above maximum ambient for high temperature are such that uncovering the core is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip settings of approximately 300 percent for high flow or 40°F above maximum ambient for temperature are based on the same criteria as the HPCI.

The reactor water cleanup system high temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that uncovering the core is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to the Safety Limit. The trip

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3.5 (cont'd)

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

H. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of axial location and average planar exposure shall be within limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown in Figures 3.5-11 through 3.5-14 during two recirculation loop operation. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by 0.84 (see Bases 3.5.K, Reference 1). If anytime during reactor power operation greater than 25% of rated power it is determined that the limiting value for APLHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, an orderly reactor power reduction shall be commenced immediately. The reactor power shall be reduced to less than 25% of rated power within the next four hours, or until the APLHGR is returned to within the prescribed limits.

4.5 (cont'd)

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

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

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

H. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25% rated thermal power.

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3.5 BASES

A. Core Spray System and Low Pressure Coolant injection (LPCI) Mode of the RHR System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

The loss-of-coolant analysis if referenced and described in General Electric Topical Report NEDE-24011-P-A.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems. No single failure of ECCS equipment occurring during a loss-of-coolant accident under these limiting conditions of operation will result in inadequate cooling of the reactor core.

Core spray distribution has been shown, in full scale tests of systems similar in design to that of the FitzPatrick Plant, to exceed the minimum requirements by at least 25 percent. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor before the internal pressure has fallen to 113 psi above primary containment pressure.

The LPCI mode of the RHR System is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the Core Spray System; however, it does function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI mode of

3.6 (cont'd)

2.

a. From and after the date that the safety valve function of one safety/relief valve is made or found to be inoperable, continued operation is permissible only during the succeeding 30 days unless such valve is made operable sooner.

b. From and after the time that the safety valve function on two safety/relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 7 days unless such valves are sooner made operable.

3. If Specification 3.6.E.1 and 3.6.E.2 are not met, the reactor shall be placed in a cold condition within 24 hours.

4. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in Item B.2 above, provided that reactor coolant temperature is $\leq$212°F and the reactor vessel is vented or the reactor vessel head is removed.

4.6 (cont'd)

2. At least one safety/relief valve shall be disassembled and inspected once/operating cycle.

3. The integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves shall be demonstrated at least once every 3 months.

4. An annual report of safety/refief vave failures and challenges will be sent to the NRC in accordance with Section 6.9.A.2.b.

LIMITING CONDITIONS FOR OPERATION

3.12 FIRE PROTECTION SYSTEMS

Applicability:

Applies to the Operational Status of the Fire Protection Systems.

Objective:

To assure operability of the Fire Protection Systems.

Specification:

A. High Pressure Water Fire Protection System

1.

a. Both high pressure water fire protection pumps and associated automatic and manual initiation logic shall be operable and aligned to the high pressure water fire header.

b. The high pressure water fire protection system shall be operable with an operable flow path capable of taking suction from the lake and transferring the water through distribution piping with operable sectionalizing control or isolation valves to the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler, or spray system riser required to be operable per specification 3.12.B and to each hose station riser isolation valve required to be operable per specification 3.12.D.

SURVEILLANCE REQUIREMENTS

4.12 FIRE PROTECTION SYSTEMS

Applicability:

Applies to the Surveillance of the Fire Protection System.

Objective:

To verify the operability of the Fire Protection Systems.

Specification:

A. High Pressure Water Fire Protection System

1. High pressure water fire protection system testing:

	Item	Frequency
a.	High pressure water fire protection system pressure check.	Once/week
b.	Each pump, on a STAGGERED TEST BASIS, by starting and operating it for at least 20 minutes on recirculating flow	Once/month
c.	Valve operational test	Once/12 months
d.	System flush	Once/6 months
e.	Functional test including:	Once/18 months

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6.0 ADMINISTRATIVE CONTROLS

Administrative Controls are the means by which plant operations are subject to management control. Measures specified in this section provide for the assignment of responsibilities, plant organization, staffing qualifications and related requirements, review and audit mechanisms, procedural controls and reporting requirements. Each of these measures are necessary to ensure safe and efficient facility operation.

6.1 RESPONSIBILITY

The Resident Manager is responsible for safe operation of the plant. During periods when the Resident Manager is unavailable, the Superintendent of Power will assume his responsibilities. In the event both are unavailable, the Resident Manager may delegate this responsibility to other qualified supervisory personnel. The Resident Manager reports directly to the Executive Vice President-Nuclear Generation.

6.2 ORGANIZATION

6.2.1 Facility Management and Technical Support

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities that affect the safety of the nuclear power plant.

1. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Updated FSAR.
2. The Resident Manager shall be responsible for overall plant operation, and shall have control over those onsite activities that are necessary for safe operation and maintenance of the plant.
3. The Executive Vice President - Nuclear Generation shall take any measures needed to ensure acceptable performance of the stall in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.
4. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Plant Staff

The plant staff organization shall be as follows:

1. Each shift crew shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

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2. An SRO or SRO with a license limited to fuel handling shall directly supervise all Core Alternations. This person shall directly supervise all Core Alterations. This person shall have no other duties during this time;

3. 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;

4. In the event of illness or unexpected absence, up to two (2) hours is allowed to restore the shift crew or fire brigade to the minimum compliment;

5. The Operations Superintendent, Assistant Operations Superintendent, Shift Supervisor and Assistant Shift Supervisor shall hold a SRO license and the Senior Nuclear Operator and the Nuclear Control Operator shall hold a RO license.

6. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and maintenance personnel who are working on safety-related systems.

   Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating.

   However, in the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance or major modifications, on a temporary basis, the following guidelines shall be followed:

   a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

   b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

   c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

   d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

   Any deviation from the above guidelines shall be authorized by the Resident Manager or the Superintendent of Power, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Resident Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

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6.3 PLANT STAFF QUALIFICATIONS

6.3.1 The minimum qualifications with regard to educational background and experience for plant staff positions shown in FSAR Figure 13.2-7 shall meet or exceed the minimum qualifications of ANSI N18.1-1971 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.

6.3.2 The Shift Technical Advisor (STA) shall meet or exceed the minimum requirements of either Option 1 (Combined SRO/STA Position) or Option 2 (Continued use of STA Position), as defined in the Commission Policy Statement on Engineering Expertise on Shift, published in the October 28, 1985 Federal Register (50 FR 43621). When invoking Option 1, the STA role may be filled by the Shift Supervisor or Assistant Shift Supervisor. (1)

6.3.3 Any deviations will be justified to the NRC prior to an individual's filling of one of these positions.

NOTE:

(1) The 13 individuals who hold SRO licenses, and have completed the FitzPatrick Advanced Technical Training Program prior to the issuance of License Amendment 111, shall be considered qualified as dual-role SRO/STAs.

6.4 RETRAINING AND REPLACEMENT TRAINING

A training program shall be maintained under the direction of the Training Superintendent 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 N18.1-1971.

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 21-1975, 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 separate groups for plant operations have been constituted. One of these, the Plant Operating Review Committee (PORC), is an onsite review group. The other is an independent review and audit group, the offsite Safety Review Committee (SRC).