

BEFORE THE UNITED STATES
NUCLEAR REGULATORY COMMISSION

In the Matter of)
POWER AUTHORITY OF THE STATE OF NEW YORK) Docket No. 50-333
(James A. FitzPatrick Nuclear Power Plant))

APPLICATION FOR AMENDMENT
TO
OPERATING LICENSE

The Power Authority of the State of New York, Licensee in the above-captioned docket, hereby files an Application for Amendment to Operating License DPR-59, which would make certain changes to the Technical Specifications as set forth in Appendix A.

With this Application for Amendment, Licensee hereby transmits documents entitled "Proposed Changes to Technical Specifications" (Attachment A) and "James A. FitzPatrick Nuclear Power Plant Pump and Valve Inservice Testing Program", Revision 1, January 6, 1979 (Attachment B). The proposed changes would provide for revisions to Sections 1.0, 4.5, 4.7 and 4.11 of the Specifications. The proposed changes would delete explicit requirements for operability testing of certain pumps and valves from the above Specification Sections 4.5, 4.7 and 4.11, and instead apply the Pump and Valve Operability Test Program for FitzPatrick (Attachment B) to the Specifications.

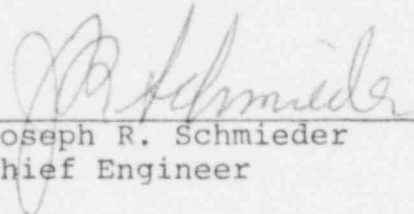
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The proposed changes would not authorize any change in the types or any increase in the amounts of effluents or any increase in the authorized power level of the facility.

WHEREFORE, Applicant respectfully requests that Appendix A to Facility Operating License No. DPR-59 be amended in the form attached hereto as Attachment A.

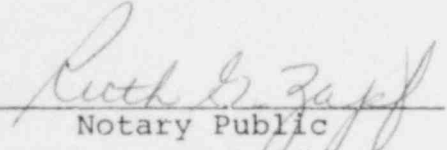
POWER AUTHORITY OF THE STATE
OF NEW YORK

By



Joseph R. Schmieder
Chief Engineer

Subscribed and sworn to
before me this 9 day
of March, 1979.



Notary Public

RUTH G. ZAPP
Notary Public, State of New York
No. 30-4663428
Qualified in Nassau County
Commission Expires March 30, 1980

ATTACHMENT A

Power Authority of the State of New York

License No. DPR-59

Docket No. 50-333

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

1.0 (cont'd)

opened to perform necessary operational activities.

2. At least one door in each air-lock is closed and sealed.
3. All automatic containment isolation valves are operable or de-activated in the isolated position.
4. All blind flanges and manways are closed.

N. Rated Power - Rated power refers to operation at a reactor power of 2,436 MWt. This is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated nuclear system pressure, refer to the values of these parameters when the reactor is at rated power.

O. Reactor Power Operation - Reactor power operation is any operation with the Mode Switch in the Startup/Hot Standby or Run position with the reactor critical and above 1 percent rated thermal power.

P. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space sensor.

Q. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the Plant subsequent to that refueling.

R. Safety Limits - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

S. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.
2. The Standby Gas Treatment System is operable.
3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

1.0 (Cont'd)

- T. Surveillance Frequency - Periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted \pm 25 percent. The interval as pertaining to instrument and electrical surveillance shall never exceed one operating cycle. In cases where the elapsed interval has exceeded 100 percent of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.
- U. Surveillance Requirements - Inservice inspection and testing of ASME Code Class 1, 2 and 3 pumps and valves shall be applicable as follows:
1. Inservice inspection of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and Applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).
 2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code

and applicable Addenda shall be applied as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities.	Required frequencies for performing inservice inspection and testing activities.
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Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

3. The provisions of Specification 1.0.T, the definition of Surveillance Frequency, are applicable to the above required frequencies for performing inservice inspection and testing activities.
4. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.
5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

1.0 (cont'd)

V. Thermal Parameters

1. **Minimum critical power ratio (MCPR) -** Ratio of that power in a fuel assembly which is calculated to cause some point in that fuel assembly to experience boiling transition to the actual assembly operating power as calculated by application of the GEXL correlation (Reference NEDE-10958).
2. **Fraction of Limiting Power Density -** The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type. Design LHGR's are 18.5 KW/ft for 7x7 bundles and 13.4 KW/ft for 8x8 and 8x8R bundles.
3. **Maximum Fraction of Limiting Power Density -** The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).
4. **Transition Boiling -** Transition boiling means the boiling region between nucleate and film boiling. Transition boiling is the region in which both nucleate and film boiling occur intermittently with neither type being completely stable.

W. Electrically Disarmed Control Rod

To disarm a rod drive electrically, the four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred. Electrical disarming does not eliminate position indication.

X. High Pressure Water Fire Protection System

The High Pressure Water Fire Protection System consists of: a water source and pumps; and distribution system piping with associated post indicator valves (isolation valves). Such valves include the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler or water spray subsystem.

Y. Staggered Test Basis

A Staggered Test Basis shall consist of:

- a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.
- b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

3.5 (cont'd)

4.5 (cont'd)

- | | |
|---|--------------------------------|
| b. Flow Rate Test -
Core spray pumps
shall deliver at
least 4,625 gpm
against a system
head corresponding
to a total pump
developed head of
≥ 113 psig | Once/3 months |
| c. Logic System
Functional Test | Once/each opera-
ting cycle |
| d. Core Spray Header
Δp Instrumentation | |
| Check | Once/day |
| Calibrate | Once/3 months |
| Test | Once/3 months |

3.5 (cont'd)

2. From and after the date that one of the Core Spray Systems is made or found inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the system is made operable earlier, provided that during the 7 days all active components of the other Core Spray System and the LPCI System and the emergency diesel generators shall be operable.
3. The LPCI mode of the RHR System shall be operable whenever irradiated fuel is in the reactor and prior to reactor startup from a cold condition, except as specified below.
 - a. From the time that one of the RHR pumps is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the pump is made operable earlier provided that during such 7 days the remaining active components of the LPCI, containment spray mode, all active components of both Core Spray Systems, and the emergency diesel generators are operable.

4.5 (cont'd)

2. When it is determined that one Core Spray System is inoperable, the operable Core Spray System, the LPCI System, and the emergency diesel generators shall be demonstrated to be operable immediately. The remaining Core Spray System shall be demonstrated to be operable daily thereafter.
3. LPCI System testing shall be as specified in 4.5.A.1.a, b and c, except that three RHR pumps shall deliver at least 23,100 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.
 - a. When it is determined that one of the RHR pumps is inoperable, the remaining active components of the LPCI, containment spray subsystem, both Core Spray Systems, and the emergency diesel generators required for operation shall be demonstrated to be operable immediately and the remaining RHR pumps shall be demonstrated to be operable daily thereafter.

3.5 (cont'd)

5. All recirculation pump discharge valves and bypass valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).

6. If the requirements of 3.5.A cannot be met, the reactor shall be placed in the cold condition within 24 hr.

B. Containment Cooling Subsystem Mode (of the RHR System)

1. Both subsystems of the containment cooling mode, each including two RHR, one ESW pump and two RHRSW pumps shall be operable whenever there is irradiated fuel in the reactor

4.5 (cont'd)

5. All recirculation pump discharge and bypass valves shall be tested for operability any time the reactor is in the cold condition exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.

B. Containment Cooling Subsystem Mode (of the RHR System)

1. Subsystems of the containment cooling mode are tested in conjunction with the test performed on the LPCI System and given in 4.5.A.1.a and b. Residual heat removal

3.5 (cont'd)

4. Should one of the containment cooling subsystems become inoperable, continued reactor operation is permissible for a period not to exceed 7 days, unless such subsystem is sooner made operable provided that during such 7 days all active components of the other containment cooling subsystem, including its associated diesel generator, are operable.
5. If the requirements of 3.5.B cannot be met, the reactor shall be placed in a cold condition within 24 hrs.
6. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature $\leq 212^{\circ}\text{F}$ with an inoperable component(s) as specified in 3.5.B above.

C. High Pressure Coolant Injection (HPCI) System

1. The HPCI System shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition, except as specified below:

4.5 (cont'd)

C. High Pressure Coolant Injection (HPCI) System

Surveillance of HPCI System shall be performed as follows provided a reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within ten days of continuous operation from the time steam becomes available.

1. HPCI System testing shall be as specified in 4.5.A.1.a, b and c except that the HPCI pump shall deliver at least 4,250 gpm against a system head corresponding to a reactor vessel pressure of 1,120 psig to 150 psig.

3.5 (cont'd)

E. Reactor Core Isolation Cooling (RCIC) System

1. The RCIC System shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 150 psig and prior to a reactor startup from a cold condition, except from the time that the RCIC System is made or found to be inoperable for any reason, continued reactor power operation is permissible during the succeeding 7 days unless the system is made operable earlier provided that during these 7 days the HPCI System is operable.
2. If the requirements of 3.5.E cannot be met, the reactor shall be placed in the cold condition and pressure less than 150 psig within 24 hours.
3. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in 3.5.E.2 above, provided that reactor coolant temperature is $\leq 212^{\circ}\text{F}$.

4.5 (cont'd)

E. Reactor Core Isoiation Cooling (RCIC) System

1. RCIC System testing shall be performed as follows provided a reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within ten days of continuous operation from the time steam becomes available.

<u>Item</u>	<u>Frequency</u>
a. Simulated Automatic Actuation Test	Once/operating cycle
b. Flow Rate	Once/3 months

The RCIC pump shall deliver at least 400 gpm for a system head corresponding to a reactor pressure of 1,120 psig to 150 psig.

4.5 BASES

The testing interval for the Core and Containment Cooling Systems is based on a quantitative reliability analysis, industry practice, judgement, and practicality. The Emergency Core Cooling Systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 450 psig; thus, during operation even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems will be automatically actuated during a refueling outage. In the case of the Core Spray System, condensate storage tank water will be pumped to the vessel to verify the operability of the core spray header. To increase the availability of the individual components of the Core and Containment Cooling Systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Pumps and motor-operated valves are tested to assure their operability in accordance with the ASME Section XI Pump and Valve

Test Program for FitzPatrick. The combination of automatic actuation test and pump and valve operability tests is deemed to constitute adequate testing of these systems.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The surveillance requirements to ensure that the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC Systems are filled provides for a visual observation that water flows from a high point vent. This ensures that

the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided in the Core Spray System and LPCI System to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the interval between the monthly checks the status of the discharge piping is monitored on a continuous basis.

3.7 (cont'd)

D. Primary Containment Isolation Valves

1. During reactor power operating conditions, all isolation valves listed in Table 3.7-1 and all instrument line flow check valves shall be operable, except as specified in 3.7.D.2.

4.7 (cont'd)

- c. Secondary containment capability to maintain a 1/4 in. of water vacuum under calm wind conditions with a filter train flow rate of not more than 6,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:
 - a. At least once per operating cycle, the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.
 - b. At least once per operating cycle, the instrument line excess flow check valves shall be tested for proper operation.

3.7 (cont'd)

4.7 (cont'd)

2. In the event any isolation valve specified in Table 3.7-1 becomes inoperable, reactor power operation may continue, provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.
3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold condition within 24 hr.

2. Whenever an isolation valve listed in Table 3.7-1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

by in-place testing with DOP as testing medium.

The test interval for filter efficiency was selected to minimize plugging of the filters. In addition, retention capacity in terms of milligrams of iodine per gram of charcoal will be demonstrated. This will be done by testing the charcoal once a year, unless filter efficiency seriously deteriorates. Since shelf lives greater than 5 yr. have been demonstrated, the test interval is reasonable.

D. Primary Containment Isolation Valves

The large pipes comprising a portion of the Reactor Coolant System, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except those lines needed for Emergency Core Cooling Systems operation or containment cooling). The closure times specified herein are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, isolation valve closure time is sufficient to prevent uncovering

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 sec.

For Reactor Coolant System temperatures less than 212°F, the containment could not become pressurized due to a loss-of-coolant accident. The 212°F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels.

The primary containment isolation valves are highly reliable, have low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate.

4.7 BASES (cont'd)

The main steam line isolation valves and all other normally open power-operated isolation valves are functionally tested in accordance with the ASME Section XI Pump and Valve Test Program for FitzPatrick.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 in. restricting orifice inside the primary containment and an excess flow check valve outside the primary containment.

3.11 (Cont'd)

D. Emergency Service Water System

1. To ensure the adequate equipment and area cooling, both ESW systems shall be operable when the requirements of specification 3.5.A and 3.5.B must be satisfied, except as specified below in specification 3.11.D.2.

4.11 (Cont'd)

D. Emergency Service Water System

1. Surveillance of the ESW system shall be performed as follows:

	<u>Item</u>	<u>Frequency</u>
a.	Simulated Automatic Actuation Test	Each operating cycle
b.	Flow Rate Test - ESW pumps shall deliver at least 3,700 gpm against a system head corresponding to a total pump head of ≥ 80 psi, as determined from the pump certification curve by measuring the pump shutoff head which shall be ≥ 120 psi.	Once/3 months
c.	Instrumentation test	Once/3 months

3.11 (Cont'd)

2. From and after the time that one Emergency Service Water System is made or found to be inoperable for any reason continued reactor operation is permissible for a period not to exceed 7 days total for any calendar month, provided that: the operable Emergency Diesel Generator System and all its emergency loads be demonstrated to be operable immediately and daily thereafter.
3. If specification 3.11.D.2 cannot be met, an orderly shut-down shall be initiated and the reactor shall be placed in a cold condition within 24 hours.

4.11 (Cont'd)

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|----|------------------------------------|---------------------------------|
| d. | Instrumentation
Calibration | Once/each
operating
cycle |
| e. | Logic System
Functional
Test | Once/each
operating
cycle |
2. ESW will not be supplied to RBCLC system during testing.

3.11 & 4.11 BASESA. Main Control Room Ventilation System

One main control room emergency ventilation air supply fan provides adequate ventilation flow under accident conditions. Should one emergency ventilation air supply fan and/or fresh air filter train be out of service during reactor operation, the allowable repair time of 1 month is justified, based on the 3 month test interval.

The 3 month test interval for the main control room emergency ventilation air supply fan and dampers is sufficient since two redundant trains are provided and neither is normally in operation.

A pressure drop test across each filter and across the filter system is a measure of filter system condition. DOP injection measures particulate removal efficiency of the high efficiency particulate filters. A Freon-112 test of the charcoal filters is essentially a leakage test. Since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodine, the test also gives an indication of the relative efficiency of the installed system. Laboratory analysis of a sample of the charcoal filters positively demonstrates halogen removal efficiency. These tests are

conducted in accordance with manufacturers' recommendations.

B. Crescent Area Ventilation

Engineering analyses indicate that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured.

C. Battery Room Ventilation

Engineering analyses indicate that the temperature rise and hydrogen buildup in the battery, and battery charger compartments without adequate ventilation is such that continuous operation of equipment in these compartments cannot be assured.

D. Emergency Service Water System

The ESWS has two 100 percent cooling capacity pumps, each powered from a separate standby power supply. The ESWS utilizes lake water to the cooling system of the emergency diesel generators. The system will also supply water to those components of the RBCLCS which are required for emergency conditions during a loss of power condition. These include ECCS pumps and area unit coolers. Pump and valve functional testing is performed in accordance with the ASME Section XI Pump and Valve Operability Test Program for FitzPatrick.