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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. 73345)

The Commission has issued the enclosed Amendment No. 148 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 May 31, 1989, amended by letter dated July 18, 1989, and amplified by letter dated November 20, 1989.

The amendment modifies the Residual Heat Removal surveillance criteria to reflect deletion of the Low Pressure Coolant Injection System loop selection logic scheme. It also clarifies the use of the terms "demonstrate" and "verify" throughout the Technical Specifications.

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. 148 to DPR-59
2. Safety Evaluation

cc: w/enclosures
See next page

| OFC | :LA:PDI-1 | :PM:PDI-1 | :SRXB | :OTSB | :PD:PDI-1 | : OGC | : |
|---|---|---|---|---|---|---|---|
| NAME | :CVogan | :DLaBarge/bd/bah | :RJones | :JCalvo | :RCapra | : BB | : |
| DATE | :11/21/89 | :11/21/89 | :11/ /89 | :11/30/89 | :12/26/89 | : 12/6 /89 | : |

Done
12/26

OGC concurrence conditioned on Amendment not Being issued until After (12-11 - Expiration date for 2d Weekly notice) And Insertion of noted Language in SE

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Document Name: AMEND TAC 73345

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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. 148
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 May 31, 1989, amended by letter dated July 18, 1989 and amplified by letter dated November 20, 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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 148, 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

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: December 26, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 148

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
| 89 | 89 |
| 106 | 106 |
| 109 | 109 |
| 114 | 114 |
| 115 | 115 |
| 115a | 115a |
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| 241 | 241 |

3.3 (cont'd)

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 started 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 started until the affected control rod drive has been replaced or repaired.

4.3 (cont'd)

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. A second licensed operator shall verify the conformance to Specification 3.3.A.2.d before a rod may be bypassed in the Rod Sequence Control System.

d. Once per week check status of pressure and level alarms for each accumulator.

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A shall be considered fulfilled, and continued operation permitted, provided that:

1. The component is returned to an operable condition within 7 days.

4.4 (cont'd)

pump solution in the recirculation path.

Explode one of three primer assemblies manufactured in same batch to verify proper function. Then install the two remaining primer assemblies of the same batch in the explosive valves.

Demineralized water shall be injected into the reactor vessel to test that valves (except explosive valves) not checked by the recirculation test are not clogged.

Test that the setting of the system pressure relief valves is between 1,400 and 1,490 psig.

3. Disassemble and inspect one explosive valve so that it can be established that the valve is not clogged. Both valves shall be inspected in the course of two operating cycles.

B. Operation with Inoperable Components

When a component becomes inoperable its redundant component shall be verified to be operable immediately and daily thereafter.

ATWS requirements are satisfied at all concentrations above 10 weight percent for a minimum enrichment of 34.7 atom percent of B-10.

Figure 3.4-1 shows the permissible region of operation on a sodium pentaborate solution volume versus concentration graph. This curve was developed for 34.7% enriched B-10 and a pumping rate of 50 gpm. Each point on this curve provides a minimum of 660 ppm of equivalent natural boron in the reactor vessel upon injection of SLC solution. At a solution volume of 2200 gallons, a weight concentration of 13% sodium pentaborate, enriched to 34.7% boron-10 is needed to meet shutdown requirements. The maximum storage volume of the solution is 4780 gallons which is the net overflow volume in the SLC tank.

Boron concentration, isotopic enrichment of boron-10, solution temperature, and volume are checked on a frequency adequate to assure a high reliability of operation of the system should it every be required. Experience with pump operability indicates that monthly testing is adequate to detect if failures have occurred.

The only practical time to test the Standby Liquid Control System is during a refueling outage and by initiation from local stations. Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of more than once each refueling outage unnecessary. A test of explosive charges from one manufacturing batch is made to assure that the charges are satisfactory. A continuous check of the firing circuit continuity is provided by pilot lights in the control room.

The relief valves in the Standby Liquid Control System protect the system piping and positive displacement pumps, which are nominally designed for 1,500 psig, from overpressure. The pressure relief valves discharge back to the standby liquid control pump suction line.

B. Operation with Inoperable Components

Only one of two standby liquid control pumping circuits is needed for operation. If one circuit is inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue during repairs. Assurance that the remaining system will perform its function is obtained by verifying pump operability in the operable circuit at least daily.

C. Sodium Pentaborate Solution

To guard against precipitation, the solution, including that in the pump suction piping, is kept at least 10°F above saturation temperature. Figure 3.4-2 shows the saturation temperature including 10°F margin as a function of sodium pentaborate solution concentration. Tank heater and heat tracing system are provided to assure compliance with this requirement. The set points for the automatic actuation of the tank heater and heat tracing system are established based on the solution concentration. Temperature and liquid level alarms for the system annunciate in the control room. Pump operability is checked on a frequency to assure a high reliability of operation of the system should it ever be required.

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 shall be operable.

3. Both LPCI subsystems 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 LPCI subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless that subsystem is made operable earlier provided that during these 7 days the operable LPCI subsystem and both Core Spray Systems shall be operable.

4.5 (cont'd)

2. When it is determined that one Core Spray System is inoperable, the operable Core Spray System, and both LPCI subsystems, shall be verified to be operable immediately. The remaining Core Spray System shall be verified to be operable daily thereafter.

3. LPCI System testing shall be as specified in 4.5.A.1a, b, c, d, f and g except that each RHR pump shall deliver at least 9,900 gpm against a system head corresponding to a reactor vessel to primary containment differential pressure of greater than or equal to 20 psid.

   a. When it is determined that one LPCI subsystem is inoperable, the operable LPCI subsystem and both Core Spray Systems shall be verified to be operable immediately and daily thereafter.

3.5 (cont'd)

b. When the reactor water temperature is greater than 212°F, the motor operator for the RHR cross-tie valve (MOV20) shall be maintained disconnected from its electric power source. It shall be maintained chain-locked in the closed position. The manually operated gate valve (10-RHR-09) in the cross-tie line, in series with the motor operated valve, shall be maintained locked in the closed position.

4. a. The reactor shall not be started up with the RHR System supplying cooling to the fuel pool.

b. The RHR System shall not supply cooling to the spent fuel pool when the reactor coolant temperature is above 212°F.

4.5 (cont'd)

b. The power source disconnect and chain lock to motor operated RHR cross-tie valve, and lock on manually operated gate valve shall be inspected once each operating cycle to verify that both valves are closed and locked.

3.5 (cont'd)

5. All recirculation pump discharge 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 hrs.

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 vessel, prior to startup from a cold condition, and reactor coolant temperature $\geq$212°F except as specified below:

2. Continued reactor operation is permissible for 30 days with one spray loop inoperable and with reactor water temperature greater than 212°F.

4.5 (cont'd)

5. All recirculation pump discharge 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 subsystems and given in 4.5.A.1.a, b, c, and d. Residual heat removal service water pumps, each loop consisting of two pumps operating in parallel, will be included in testing, supplying 8,000 gpm. The Emergency Service Water System, each loop of which consists of a single operating emergency service water pump will be tested in accordance with Section 4.11D.

   During each five-year period, an air test shall be performed on the containment spray headers and nozzles.

2. When it is determined that one RHR pump and/or one RHRSW pump of the components required in 3.5.B.1 above are inoperable, the remaining redundant active components of the containment cooling mode subsystems shall be verified to be operable immediately and daily thereafter.

3.5 (cont'd)

3. Should one RHR pump and/or one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining active components of the containment cooling mode are operable.

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 are operable.

5. If the requirements of 3.5.B cannot be met, the reactor shall be placed in a cold condition within 24 hr.

6. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature <212°F with an inoperable component(s) as specified in 3.5.B above.

4.5 (cont'd)

3. When one containment cooling subsystem loop becomes inoperable, the operable loop shall be verified to be operable immediately and daily thereafter.

3.5 (cont'd)

a. From and after the date that the HPCI System is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the Automatic Depressurization System, the Core Spray System, LPCI System, and Reactor Core Isolation Cooling System are operable.

b. If the requirements of 3.5.C.1 cannot be met, the reactor shall be placed in the cold condition and pressure less than 150 psig within 24 hrs.

2. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature $\leq$212°F with an inoperable component(s) as specified in 3.5.C.1 above.

4.5 (cont'd)

a. When it is determined that the HPCI subsystem is inoperable the RCIC, the LPCI subsystem, both core spray subsystems, and the ADS subsystem actuation logic shall be verified to be operable immediately. The RCIC system and ADS subsystem logic shall be verified to be operable daily thereafter.

3.5 (cont'd)

during such time, the HPCI System is operable.

2. If the requirements of 3.5.D.1 cannot be met, the reactor shall be placed in the cold condition and pressure less than 100 psig, within 24 hr.

3. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in 3.5.1.a and 3.5.1.b above, provided that reactor coolant temperature is $\leq$212°F and the reactor vessel is vented or reactor vessel head is removed.

4.5 (cont'd)

2. A logic system functional test.

a. When it is determined that one valve of the ADS is inoperable, the ADS subsystem actuation logic for the operable ADS valves and the HPCI subsystem shall be verified to be operable immediately and at least weekly thereafter.

b. When it is determined that more than one relief/safety valve of the ADS is inoperable, the HPCI System shall be verified to be operable immediately.

3.5 (cont'd)

4.5 (cont'd)

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.

2. When it is determined that the RCIC System is inoperable at a time when it is required to be operable, the HPCI System shall be verified to be operable immediately and daily thereafter.

## 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 is 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 psig.

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. These subsystems are completely independent of the Core Spray System; however, they function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI mode of

3.5 BASES (cont'd)

the RHR System in conjunction with the Core Spray System provides adequate cooling for break areas of approximately 0.2 sq. ft. up to and including the double-ended reactor recirculation line break without assistance from the high pressure Emergency Core Cooling Systems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference 8. Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the Core Spray and LPCI Systems constitute 1-out-of-2 systems; however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 30 days, and this specification is within this period. For multiple failures, a shorter interval is specified and to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in Reference 8 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgement.

Should one Core Spray System become inoperable, the remaining Core Spray and the entire LPCI System are available should the need for core cooling arise. To assure that the remaining Core Spray and LPCI Systems are available, they are verified operable immediately. This verification includes the pumps and associated valves. Based on judgements of the reliability of the remaining systems, i.e., the Core Spray and LPCI, a seven-day repair period was obtained. Similarly, should one LPCI subsystem become inoperable, the remaining subsystem and the Core Spray System are available to provide cooling.

3.5 BASES (cont'd)

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

The containment heat removal portion of the LPCI/containment spray mode is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and, therefore, is more than ample to provide the required heat removal capability.

The containment cooling mode (of the RHR System) consists of two sets of two RHR Pumps, two RHR service water pumps, one ESW Pump, and one heat exchanger. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability as any two of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a thirty-day repair period is adequate. Loss of one subsystem of the containment cooling mode leaves one remaining system to perform the containment cooling function. The operable system is verified to be operable each day when the above condition occurs. Based on the fact that when one containment cooling subsystem becomes inoperable only one system remains, a seven day repair period was specified.

Low power physics testing and reactor operator training with inoperable components will be conducted only when the containment cooling mode of RHR is not required for the safety of the plant.

Calculations have been made to determine the effects of the design basis LOCA while conducting low power physics testing or operator training at or below 212°F. The results of these conservative calculations show that the suppression pool water temperature will not exceed 170°F. Therefore LPCI and Core Spray Systems will not be adversely affected by the postulated LOCA.

3.5 BASES (cont'd)

vessel head off the LPCI and Core Spray Systems will perform their designed safety function without the help of ADS.

E. Reactor Core Isolation Cooling (RCIC) System

The RCIC is designed to provide makeup to the Reactor Coolant System as a planned operation for periods when the normal heat sink is unavailable. The RCIC also serves as redundant makeup system on total loss of all offsite power in the event that HPCI is unavailable. In all other postulated accidents and transients, the ADS provides redundancy for the HPCI. Based on this and judgements on the reliability of the HPCI system, an allowable repair time of 7 days is specified. Immediate and daily verifications of HPCI operability during RCIC outage is considered adequate based on judgement and practicality.

Low power physics testing and reactor operator training with inoperable components will be conducted only when the RCIC System is not required, (reactor coolant temperature $\leq$212°F and coolant pressure $\leq$150 psig). If the plant parameters are below the point where the RCIC System is required, physics testing and operator training will not place the plant in an unsafe condition.

Operability of the RCIC System is required only when reactor pressure is greater than 150 psig and reactor coolant temperature is greater than 212°F because core spray and low pressure coolant injection can protect the core for any size pipe break at low pressure.

F. Minimum Emergency Core and Containment Cooling System Availability

The purpose of Specification 4.5.D is to assure a minimum of emergency core cooling equipment is available at all times. If, for example, one core spray were out of service and the emergency bus which powered the opposite core spray were out of service, only two RHR Pumps would be available. Likewise, if two RHR pumps were out of service and two RHR on the opposite side were also out of service, no containment cooling would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the Reactor Coolant System which could lead to draining the vessel. This work would include work on certain control rod drive components and Reactor Recirculation System. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other

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. Likewise, the pumps and motor-operated valves are also tested each month to assure their operability. The combination automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service, overall core and containment cooling reliability is maintained by verifying the operability of the remaining cooling equipment. Consistent with the definition of operable in Section 4.0.C, demonstrate means conduct a test to show; verify means that the associated surveillance activities have been satisfactorily performed within the specified time interval.

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

3.6 (cont'd)

which are required to be operable in these modes, complete one of the following:

a. replace or restore the inoperable snubber(s) to operable status or,

b. declare the supported system inoperable and follow the appropriate limiting condition for operation statement for that system or,

c. perform an engineering evaluation to show the inoperable snubber is unnecessary to assure operability of the system or to meet the design criteria of the system, and remove the snubber from the system.

3. With one or more snubbers found inoperable, within 72 hours perform a visual inspection of the supported component(s) associated with the inoperable snubber(s) and document the results. For all modes of operation except Cold Shutdown and Refueling, within 14 days complete an engineering evaluation as per Specification 4.6.I.6 to ensure that the inoperable snubber(s) has not adversely affected the supported component(s). For Cold Shutdown or Refueling mode, this evaluation shall be completed within 30 days.

4.6 (cont'd)

# The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

2. Visual inspection shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movements can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.6.I.7 or 4.6.I.8, as applicable. Hydraulic snubbers which have lost sufficient fluid to potentially cause uncovering of the fluid reservoir-to-snubber valve assembly port or bottoming of the fluid reservoir piston with the snubber

3.6 and 4.6 BASES (cont'd)

H. (DELETED)

I. Shock Suppressors

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation. Snubbers excluded from this inspection program are those installed on non-safety related system and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system. Because the snubber protection is required only during low probability events, a period of 72 hours (for normal operation) or 7 days (for cold shutdown or refueling mode of operation) is allowed for repairs or replacement of the snubber prior to taking any other action. Following the 72 hour (or 7 day) period, the supported system must be declared inoperable and the Limiting Condition of Operation statement for the supported system followed. As an alternative to snubber repair or replacement an engineering evaluation may be performed: to show that the inoperable snubber is unnecessary to assure operability of the system or to meet the design criteria of the system; and, to remove the snubber from the system. With one or more snubbers found inoperable, within 72 hours a visual inspection shall be performed on the supported component(s) associated with the inoperable snubber(s) and the results shall be documented. For all modes of operation except Cold Shutdown and Refueling, within 14 days an engineering evaluation shall be performed to ensure that the inoperable snubber(s) has not adversely affected the supported component(s). For Cold Shutdown or refueling mode, this evaluation shall be completed within 30 days. A period of 7 days has been selected for repair or replacement of the inoperable snubber during cold shutdown or refueling mode of operation becuase in these modes the relative probability of structural damage to the piping systems would be lower due to lower values of total stresses on the piping systems. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures.

3.7 (cont'd)

2. From and after the date that one circuit of the standby Gas Treatment System is made or found to be inoperable for any reason, the following would apply:

   a. If in Start-up/Hot Standby, Run or Hot Shutdown mode, reactor operation or irradiated fuel handling is permissible only during the succeeding 7 days unless such circuit is sooner made operable, provided that during such 7 days all active components of the other Standby Gas Treatment Circuit shall be operable.

   b. If in Refuel or Cold Shutdown mode, reactor operation or irradiated fuel handling is permissible only during the succeeding 31 days unless such circuit is sooner made operable, provided that during such 31 days all active components of the other Standby Gas Treatment Circuit shall be operable.

3. If Specifications 3.7.B.1 and 3.7.B.2 are not met, the reactor shall be placed in the cold condition and irradiated fuel handling operations and operations that could reduce the shutdown margin shall be prohibited.

4.7 (cont'd)

e. At least once per operating cycle, manual operability of the bypass valve for filter cooling shall be demonstrated.

f. Standby Gas Treatment System Instrumentation Calibration:

| | |
|---|---|
| differential pressure switches | Once/operating Cycle |

2. When one circuit of the Standby Gas Treatment System becomes inoperable, the operable circuit shall be verified to be operable immediately and daily thereafter.

3.9 Continued

C. Diesel Fuel

There will be a minimum of 64,000 gal. of diesel fuel on site for each operable pair of diesel generators.

1. From and after the time that fuel oil storage tank level instrumentation is made or found to be inoperable for any reason continued reactor operation is permissible indefinitely, provided that the level in the affected storage tank is manually measured at least once/day.

4.9 Continued

6. Once within one hour and at least once per eight hours thereafter, while the reactor is being operated in accordance with Specifications 3.9.B.1, 3.9.B.3 and 3.9.B.4, the availability of off-site power shall be assured by verifying correct breaker alignment and by verifying that the associated off-site electrical line is energized.

C. Diesel Fuel

Once a month the quantity of diesel fuel available in each storage tank shall be manually measured and compared to the reading of the local level indicators to ensure the proper operation thereof.

1. Once a month a sample of the diesel fuel in each storage tank shall be checked for quality as per the following:

| | |
|---|---|
| Flash Point - °F | 125°F min. |
| Pour Point - °F | 10°F max. |
| Water & Sediment | 0.50% max. |
| Ash | 0.5% max. |
| Distillation 90% Point | 540 min. |
| Viscosity (SSU) at 100°F | 40 max. |
| Sulfur | 1% max. |
| Copper Strip Corrosion | No. 3 max. |
| Cetane # | 35 min. |

3.11 (cont'd)

B. Crescent Area Ventilation

Crescent area ventilation and cooling equipment shall be operable on a continuous basis whenever specification 3.5.A, 3.5.B, and 3.5.C are required to be satisfied.

1. From and after the date that more than one unit cooler serving ECCS compartments in the same half of the crescent area are made or found to be inoperable, all ECCS components in that half of the crescent area shall be considered to be inoperable for purposes of specification 3.5.A, 3.5.B, and 3.5.C.

2. If 3.11.B.1 cannot be met, the reactor shall be placed in a cold condition within 24 hours.

C. Battery Room Ventilation

Battery room ventilation shall be operable on a continuous basis whenever specification 3.9.E is required to be satisfied.

1. From and after the date that one of the battery room ventilation systems is made or found to be inoperable, its associated battery shall be considered to be inoperable for purposes of specification 3.9.E.

4.11 (cont'd)

B. Crescent Area Ventilation

1. Unit coolers serving ECCS components shall be demonstrated operable once/3 months.

2. Temperature indicator controllers shall be calibrated once/operating cycle.

C. Battery Room Ventilation

Battery room ventilation equipment shall be demonstrated operable once/week.

1. When it is determined that one battery room ventilation system is inoperable, the remaining ventilation system shall be verified operable and daily thereafter.

2. Temperature transmitters and differential pressure switches shall be calibrated once/operating cycle.

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 is demonstrated to be operable immediately and daily thereafter; and,
   - all Emergency Diesel Generator System emergency loads are verified 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)

| | | |
|---|---|---|
| e. | ESW instrumentation-<br>check<br>calibrate test | Once/day<br>Once/3 months<br>Once/3 months |
| f. | Logic System Functional Test | Once/each operating cycle |

2. ESW will not be supplied to RBCLC system during testing.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO. 148 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 May 31, 1989 and amended by letter dated July 18, 1989, the Power Authority of the State of New York (PASNY or the licensee), requested changes to the Technical Specifications (TS) for the James A. FitzPatrick Nuclear Power Plant. The changes would modify the Residual Heat Removal (RHR) surveillance criteria to reflect deletion of the Low Pressure Coolant Injection (LPCI) System loop selection logic scheme. In addition, the amendment would clarify the use of the terms "demonstrate" and "verify" throughout the TS so that they are used consistently to specify the requirements of the various surveillance tests, to clarify the testing requirements, and to eliminate the need for redundant and unnecessary surveillance tests.

## DESCRIPTION

The design of the RHR system includes four pumps, divided into two loops. During conditions which indicate a loss of coolant accident (LOCA), all four pumps automatically start and valves align to the LPCI mode to inject water from the suppression chamber to flood the reactor vessel.

For the first operating cycle of the plant, the LPCI System was designed so that the cross-connect valve between the two RHR loops was maintained open. Therefore, when a LOCA signal was received the loop selection logic (using pressure transducers) determined which reactor recirculation loop was broken, prevented these LPCI injection valves from opening, and allowed the LPCI injection valves to the other (intact) loop to open. Thus, flow from all four RHR pumps was injected into the intact loop.

However, with the issuance of Appendix K to 10 CFR Part 50 in 1974, the emergency core cooling system (ECCS) acceptance criteria became more conservative. As a result, a plant modification was developed which was designed to ensure that, even with the single most limiting equipment failure (failure of a LPCI injection valve to open), the flow from two LPCI pumps would be available to reflood the vessel in the event of a LOCA. The modification involved elimination of the loop selection logic and shutting the cross-connect valve to divide the LPCI discharge into two independent loops. In addition, a closure signal was added to the recirculation pump discharge valves upon receipt of a LOCA signal.

With this arrangement, two RHR pumps per loop will discharge into their respective recirculation loop upon receipt of a LOCA signal. Even if a single LPCI injection valve fails to open, the flow from the other two LPCI pumps is available. This arrangement enhances system reliability.

Since the original LOCA analysis relative to the LPCI System assumed a minimum of three RHR pumps were operable, the TS surveillance requirement was based on three-pump operation. This requirement was stated in the TS as: "three RHR pumps must deliver at least 23,100 gpm against a system head corresponding to a reactor vessel pressure of 20 psig." Thus, the minimum flow value specified in related test procedures was 7700 gpm for each pump (33% of 23100).

The proposed TS change would specify that the minimum acceptable criteria for each pump is: "9900 gpm against a system head corresponding to a reactor vessel to primary containment differential pressure of greater than or equal to 20 psid."

A loss of coolant analysis was performed by the General Electric Company and reported in the Reload Analysis Report, NEDC-31317P, dated October 1986. It assumed that the loop selection logic was removed and each RHR loop injected independently into the recirculation system. The minimum RHR pump flows used in the analysis at a vessel pressure of 20 psid was 17,500 gpm for two pumps discharging into one loop (i.e., loss of one loop) or 19,800 gpm for two pumps discharging into two loops (i.e., loss of one pump per loop). Under either of these conditions, the report determined that the LOCA design criteria was satisfied. Therefore, the minimum acceptable flow value of 9,900 gpm (50% of 19,800) for each pump conforms to the flow assumed for the loss of coolant analysis, and is the value submitted by the licensee in this amendment.

The other change related to the RHR System would replace "psig" (pounds per square inch gauge) with "psid" (pounds per square inch differential) for the discharge pressure specified in the surveillance requirement. This is in conformance with the Reload Analysis Report which specifies the differential pressures between the suppression chamber and the reactor vessel (psid) rather than the pump discharge pressure (psig). Also, this conforms to present plant practices. Since this does not change the acceptance criteria of the test, the effect of the proposed change is to clarify its meaning.

The NRC staff agrees with the licensee that, as explained above, the changes to the RHR surveillance test acceptance criteria will ensure operability of the pump and are, therefore, acceptable.

The other changes proposed by the licensee would clarify the meaning of surveillance requirements specified in the TS by consistently using the words "demonstrate" and "verify." The proposal would eliminate the need for redundant and unnecessary surveillance tests performed to satisfy overlapping requirements and make the surveillance tests performed to ensure equipment operability more consistent with a generic letter issued on April 10, 1980 concerning use of the term "operable." The use of the term and its TS definition was reviewed and found to be acceptable in Amendment No. 83 issued on August 28, 1984.

The appropriateness of the use of the terms "demonstrate" and "verify" was evaluated throughout the TS by the licensee and the NRC staff. Where a specification requires testing at a specific frequency, or the intent is clearly to require performance of an actual test, or to determine operability of a component or system, no TS change was proposed and the word "demonstrate" is used. However, if the TS criteria is such that operability should be determined by ensuring that the associated surveillance tests have been performed with satisfactory results within the specified time interval, the term "verify" has been substituted.

For example, Specification 4.10.D.1.b, requires a shutdown margin demonstration when two control rods are withdrawn from the reactor core for maintenance. Since the intent of the requirement is to perform a test, the term "demonstrate" is retained. In contrast, if a subsystem or component is inoperable, the proposed change would delete the requirement to actually perform a test of redundant systems or equipment to prove operability if the surveillance tests have been performed within the required test interval and there is reasonable assurance that no degradation of system operability exists. Under these conditions, the term "verify" is used.

Additionally, if an engineering evaluation is used to determine operability, neither term can be clearly applied and a phrase such as "investigation has shown" is used. This affects control rod drive collect housing failure evaluation on TS page 89 and snubber operability on page 145c.

None of the proposed changes related to this issue would affect the existing normal surveillance testing requirements, nor would they affect the testing performed when equipment is returned to service from an inoperable condition, nor would they affect the In Service Testing (IST) program. Also, the term "demonstrate" was retained for all tests related to the Emergency Diesel Generators.

Other specific proposed changes related to this issue are as follows:

1. For the following, failure of a component will require that operability of redundant components be verified, rather than demonstrated:

    a. Standby Liquid Control System, Specification 4.4.B.

    b. Core Spray System, Specification 4.5.A.2

    c. LPCI Subsystem, Specification 4.5.A.3.a

    d. RHR pump or RHR Service Water Pump, Specification 4.5.B.2

    e. Containment Cooling Mode of the RHR System, Specification 4.5.B.3

    f. High Pressure Coolant Injection System, Specification 4.5.C.1.a

g. Automatic Depressurization System (ADS), Specification 4.5.D.2

h. Reactor Core Isolation Cooling System, Specification 4.5.E.2

i. Standby Gas Treatment System, Specification 4.7.B.2

j. Battery Room Ventilation, Specification 4.11.C.1

2. For the Emergency Service Water (ESW) System, the alternate testing requirements would be clarified to indicate that with one ESW System inoperable, it is the operability of the unaffected diesel generator system which must be demonstrated to be operable. A literal interpretation of the present TS would require testing of both diesel generator systems, which is not consistent with the operability criteria, nor is it possible due to the loss of the cooling water. Also, the proposed change would require verification, rather than demonstration, of the operability of the emergency loads upon loss of the ESW System.

3. To verify availability of offsite power, a proposed change to Specification 4.9.B.6 would require verification that the electrical line is energized, in addition to the present requirement to verify correct breaker alignment. This would clarify the conditions necessary to ensure that the offsite line is operable.

4. The information in the Bases Section corresponding to the proposed TS changes would be modified as necessary to reflect the proposed changes to the requirements.

5. The term "LPCI mode" would be replaced with "LPCI subsystem" in Specification 3.5.A.3.b to more clearly indicate that the requirements apply to one LPCI subsystem rather than the whole LPCI System.

Rather than the performance of a test to demonstrate operability of redundant equipment, the licensee has determined that verification is more appropriate as described above. In effect, this verification is a check to determine that the redundant equipment is <u>not inoperable</u>, rather than the establishment of test conditions which show that the equipment will operate. This is consistent with the desire to reduce the number of unnecessary challenges to the continued operability of the equipment by reducing, somewhat, the number of surveillance tests performed.

In summary, these changes which involve the use of "demonstrate" and "verify" terminology clarify the TS by improving consistency, conform to the definition of "operable," will enhance component reliability by reducing unnecessary surveillance tests, do not involve modifications to any system, and potentially improve associated systems reliability. Also, sufficient controls will be exercised to ensure that no indications which may affect operability will not be detected. For these reasons, and as explained above, the staff finds the changes acceptable.

Another proposed change involves Specification 3.5.A.3.a which addresses the condition when one RHR pump is made or found to be inoperable. Since loss of one RHR pump would render the associated LPCI Subsystem inoperable, this condition is also addressed in Specification 3.5.A.3.b which addresses the situation when one LPCI subsystem is made or found to be inoperable. Continued operation for seven days is allowed by each. Therefore, to eliminate duplication the licensee has proposed, and the NRC finds acceptable, the elimination of the present Specification 3.5.A.3.A.

## ENVIRONMENTAL CONSIDERATION

This amendment involves a change in a requirement with respect to the use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and changes to the 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 made a final no significant hazards consideration finding with respect to this amendment. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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.

Dated: December 26, 1989

PRINCIPAL CONTRIBUTOR:

D. LaBarge