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. 168
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 April 2, 1990, 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:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 168, 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

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: February 13, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 168

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
ii	ii
122	122
122a	122a
129	129
130	130
133	133
165	165
188	188

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3.5 (cont'd)

F. ECCS-Cold Condition

1. A minimum of two low pressure Emergency Core Cooling subsystems shall be operable whenever irradiated fuel is in the reactor, the reactor is in the cold condition, and work is being performed with the potential for draining the reactor vessel.

2. A minimum of one low pressure Emergency Core Cooling subsystem shall be operable whenever irradiated fuel is in the reactor, the reactor is in the cold condition, and no work is being performed with the potential for draining the reactor vessel.

3. Emergency Core Cooling subsystems are not required to be operable provided that the reactor vessel head is removed, the cavity is flooded, the spent fuel pool gates are removed, and the water level above the fuel is in accordance with Specification 3.10.C.

4. With the requirements of 3.5.F.1, 3.5.F.2, or 3.5.F.3 not satisfied, suspend core alterations and all operations with the potential for draining the reactor vessel. Restore at least one system to operable status within 4 hours or establish Secondary Containment Integrity within the next 8 hours.

4.5 (cont'd)

F. ECCS-Cold Condition

Surveillance of the low pressure ECCS systems required by 3.5.F.1 and 3.5.F.2 shall be as follows:

1. Perform a flowrate test at least once every 3 months on the required Core Spray pump(s) and/or the RHR pump(s). Each Core Spray pump shall deliver at least 4,625 gpm against a system head corresponding to a reactor vessel pressure greater than or equal to 113 psi above primary containment pressure. Each RHR pump shall deliver at least 9900 gpm against a system head corresponding to a reactor vessel to primary containment differential pressure of $\geq$ 20 psid.

2. Perform a monthly operability test on the required Core Spray and/or LPCI motor operated valves.

3. Once each shift verify the suppression pool water level is greater than or equal to 10.33 ft. whenever the low pressure ECCS subsystems are aligned to the suppression pool.

4. Once each shift verify a minimum of 324 inches of water is available in the Condensate Storage Tanks (CST) whenever the Core Spray System(s) is aligned to the tanks.

3.5 (cont'd)

G. Maintenance of Filled Discharge Pipe

Whenever core spray subsystems, LPCI subsystems, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

a. From and after the time that the pump discharge piping of the HPCI, RCIC, LPCI, or Core Spray Systems cannot be maintained in a filled

4.5 (cont'd)

G. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to, in order to assure that the discharge piping of the core spray subsystem, LPCI subsystem, HPCI, and RCIC are filled:

1. Every month prior to the testing of the LPCI subsystem and core spray subsystem, the discharge piping of these systems shall be vented from the high point, and water flow observed.

3.5 BASES (cont'd)

vessel head off the LPCI and Core Spray Systems will perform their designed safety function without the help of the 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. ECCS-Shutdown Mode

Low pressure Emergency Core Cooling Systems (ECCS) are required when the reactor is in a cold condition to ensure adequate coolant inventory makeup in case of an inadvertent draindown of the reactor vessel. Two low pressure ECCS subsystems are required operable to meet the single-failure criterion.

The low pressure ECCS subsystems consist of two CS systems, two LPCI subsystems, or a combination thereof. Each CS system consists of one motor-driven pump, associated piping, and valves. Each CS system is capable of transferring water to the reactor vessel from the suppression pool or, when the suppression pool is unavailable, the condensate storage tank. In the cold condition, each LPCI subsystem consists of one motor-driven pump, associated piping, and valves. Each LPCI subsystem is capable of transferring water from the suppression pool to the reactor vessel. Only one RHR pump is required per LPCI subsystem because of its larger flowrate compared to a Core Spray System. A LPCI subsystem operating in the shutdown cooling mode of RHR is considered operable for the ECCS function if it can be realigned manually (either remote or local) to the LPCI mode and is not otherwise inoperable. In the cold condition, the RHR system cross-tie valves are not required to be closed.

One low pressure ECCS subsystem provides sufficient vessel flooding capability to recover from an inadvertent vessel draindown. However, with only one low pressure system operable, the overall system reliability is reduced because a single-failure could render the ECCS incapable of performing its intended

3.5 BASES (cont'd)

function. Therefore, operation with the potential for draining the reactor vessel is not allowed with only one low pressure ECCS subsystem operable.

ECCS systems are not required to be operable during refueling conditions. Sufficient coolant inventory is available above the fuel to allow operator action to terminate the inventory loss prior to fuel uncovery in case of an inadvertent draindown.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI, RCIC, and HPCI are not filled, a water hammer can develop in this piping when the pump(s) are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this technical specification requires the discharge lines to be filled whenever the system is required to be operable. If a discharge pipe is not filled, the pumps the supply that line must be assumed to be inoperable for technical specification purposes. However, if a water hammer were to occur, the system would still perform it's design function.

H. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50 Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting values for APLHGR are specified in the Core Operating Limits Report. During Single Loop Operation a multiplier is applied to these values. The derivation of this multiplier can be found in Bases 3.5.K, Reference 1.

I. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation.

The LHGR shall be checked daily during reactor operation at 25% rated thermal power to determine if fuel burnup, or control rod movement, has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the ratio of local LHGR to average LHGR would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

4.5 BASES (cont'd)

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 bases.

Normally the low pressure ECCS subsystems required by Specification 3.5.F.1 are demonstrated operable by the surveillance tests in Specifications 4.5.A.1 and 4.5.A.3. Section 4.5.F specifies periodic surveillance tests for the low pressure ECCS subsystems which are applicable when the reactor is in the cold condition. These tests in conjunction with the requirements on filled discharge piping (Specification 3.5.G), and the requirements on ECCS actuation instrumentation (Specification 3.2.B), assure adequate ECCS capability in the cold condition. The water level in the suppression pool, or the Condensate Storage Tanks (CST) when the suppression pool is inoperable, is checked once each shift to ensure that sufficient water is available for core cooling.

3.7 LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The volume and temperature of the water in the pressure suppression chamber shall be maintained within the following limits whenever the reactor is critical or whenever the reactor coolant temperature is greater than 212°F and irradiated fuel is in the reactor vessel:

   a. Maximum vent submergence level of 53 inches.

   b. Minimum vent submergence level of 51.5 inches.

   The suppression chamber water level may be outside the above limits for a maximum of four (4) hours during required operability testing of HPCI, RCIC, RHR, CS, and the Suppression Chamber - Drywell Vacuum System.

   c. Maximum water temperature

      (1) During normal power operation maximum water temperature shall be 95°F.

4.7 SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary, and secondary containment systems.

Specification:

A. Primary Containment

1. The pressure suppression chamber water level and temperature shall be checked once per day. The accessible interior surfaces of the drywell and above the water line of the pressure suppression chamber shall be inspected at each refueling outage for evidence of deterioration. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

3.7 BASES (cont'd)

Using the minimum or maximum downcomer submergence levels given in the specification, containment pressure during the design basis accident is approximately 45 psig which is below the design of 56 psig. The minimum downcomer submergence of 51.5 in. results in a minimum suppression chamber water volume of 105,600 $ft.^3$. The majority of the Bodega tests (9) were run with a submerged length of 4 ft. and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. Additional JAFNPP specific analyses done in connection with the Mark I Containment-Suppression Chamber Integrity Program indicate the adequacy of the specified range of submergence to ensure that dynamic forces associated with pool swell do not result in overstress of the suppression chamber or associated structures.

The maximum temperature at the end of blowdown tested during the Humboldt Bay (10) and Bodega Bay tests was 170°F, and this is conservatively taken to be the limit for complete condensation of the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Using a 40°F rise (Section 5.2 FSAR) in the suppression chamber water temperature and a maximum initial temperature of 95°F, a temperature of 145°F is achieved, which is well below the 170°F temperature which is used for complete condensation.

For an initial maximum suppression chamber water temperature of 95°F and assuming the normal complement of containment cooling pumps (two LPCI pumps and two RHR service water pumps) containment pressure is not required to maintain adequate nat positive suction head (HPSH) for the core spray LPCI and HPCI pumps.

Limiting suppression pool temperature to 130°F during RCIC, HPCI, or relief valve operation, when decay heat and stored energy are removed form the primary system by discharging reactor steam directly to the suppression chamber assures adequate margin for a potential blowdown any time during RCIC, HPCI, or relief valve operation.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.