April 8, 1985

Docket No. 50-286

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Mr. C. A. McNeill, Jr.
Senior Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. McNeill:

The Commission has issued the enclosed Amendment No. 53 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated July 6, 1983, as supplemented December 3, 1984.

The amendment revises the Technical Specifications to provide for redundancy in decay heat removal capability in all modes of operation.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,
/s/JDNeighbors

Joseph D. Neighbors, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 53 to DPR-64
2. Safety Evaluation

cc: w/enclosures
See next page

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. DPR-64

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 July 6, 1983, as supplemented December 3, 1984, 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-64 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. 53, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 8, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 53

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1-1	3.1-1
-----	3.1-1a
3.1-3	3.1-3
3.3-5	3.3-5
------	3.3-5a
------	3.3-16a
3.8-2	3.8-2
3.8-5	3.8-5

3. LIMITING CONDI NS FOR OPERATION

For the cases where no exception time is specified for inoperable components, this time is assumed to be zero.

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System; operational components; heatup; cooldown; criticality; activity; chemistry and leakage.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps

a. When a reduction is made in the boron concentration of the reactor coolant, at least one reactor coolant pump or one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation.

b. When the reactor coolant system Tavg is greater than 350°F and electrical power is available to the reactor coolant pumps, and as permitted during special plant evolutions, at least one reactor coolant pump shall be in operation. All reactor coolant pumps may be de-energized for up to 1 hour provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

c. When the reactor coolant system Tavg is greater than 200°F and less than 350°F, and as permitted during special plant evolutions, at least one reactor coolant pump or one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation. All reactor coolant pumps may be de-energized with RHR not in service for up to 1 hour provided no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and core outlet temperature is maintained at least 10°F below saturation temperature.

d. When the reactor coolant system Tavg is less than 200°F, but not in the refueling operation condition, and as permitted during special plant evolutions, at least one residual heat removal pump (connected to the Reactor Coolant System) shall be in operation.

e. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

f. The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation.

g. If the requirements of 3.1.A.1.e and 3.1.A.1.f above cannot be satisfied, the reactor shall be brought to the hot shutdown condition within 1 hour.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power (P-7) can be removed with natural circulation only (1); hence, the requirement for one operating RCP above 350°F and two operating RCP's above 2% rated power provides a substantial safety factor. In addition, a single RCP or RHR pump (connected to the RCS) provides sufficient heat removal capability for removing decay heat.

The reactor shall not be operated at power levels above 10% rated power with less than four (4) reactor coolant loops in operation until safety analyses for less than four loop operation have been submitted by the licensee and approval for less than four loop operation at power levels above 10% rated power has been granted by the Commission. (See license condition 2.C.(3))

Each of the pressurizer code safety valves is designed to relieve 420,000 lbs. per hr. of saturated steam at the valve set point.

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load (2) without a direct reactor trip or any other control.

The requirement that 150 kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVS) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal off possible RCS leakage paths.

References

1) FSAR Section 14.1.6

2) FSAR Section 14.1.8

5. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.3 within the time periods specified in 3.3.A.4; then:

   a. If the reactor is critical, it shall be in the hot shutdown condition within four hours and the cold shutdown condition within the following 24 hours.

   b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.A.3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition using normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

6. When the reactor coolant system Tavg is greater than 200°F and less than 350°F, the following decay heat removal requirements shall be met:

   a. Two residual heat removal pumps together with their associated heat exchangers, piping, and valves shall be operable, OR

   b. A minimum of one residual heat removal pump and heat exchanger and a minimum of one reactor coolant pump and steam generator together with their associated piping and valves, shall be operable, OR

   c. A minimum of two reactor coolant pumps and two steam generators, together with their associated piping and valves, shall be operable, OR

   d. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system. Otherwise, if sufficient equipment is available, be in cold shutdown within 20 hours.

7. When the reactor coolant Tavg is less than 200°F, but not in the refueling operation condition, two residual heat removal pumps together with their associated heat exchangers, piping and valves shall be operable.

   a. With less than the above operable, initiate corrective action to return the required equipment to an operable status as soon as possible and suspend any operations which would reduce the boron concentration of the reactor coolant system.

   b. The above requirements may be suspended during maintenance, modifications, testing, inspection or repair provided that:

      1) an alternate means of decay heat removal is available and return of the system within sufficient time to prevent exceeding cold shutdown requirements is assured;

2) RCS perature and the source range (ectors are monitored hourly; and

3) no operations are permitted which would reduce the boron concentration of the reactor coolant system.

B. Containment Cooling and Iodine Removal Systems

1. The reactor shall not be brought above the cold shutdown condition unless the following requirements are met:

   a. The spray additive tank contains a minimum of 4000 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

   b. The five fan cooler-charcoal filter units and the two spray pumps, with their associated valves and piping, are operable.

2. The requirements of 3.3.B.1 may be modified to allow any one of the following components to be inoperable at one time:

During operating modes in the temperature range between 200°F and 350°F, a sufficient decay heat removal capability is provided by a reactor coolant pump with a steam generator heat sink or a residual heat removal loop. This redundancy ensures that a single failure will not result in a complete loss of decay heat removal.

During operating modes when the reactor coolant Tavg is less than 200°F, but not in the refueling operation condition, a sufficient decay heat removal capability is provided by a residual heat removal loop.

7. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

8. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for a least 120 hours. In addition, movement of fuel in the reactor before the reactor has been subcritical for equal to or greater than 365 hours will necessitate operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal adsorbers. For this case operability of the Containment Building Vent and Purge System shall be established in accordance with Section 4.13 of the Technical Specifications. In the event that more than one region of fuel (72 assemblies) is to be discharged from the reactor, those assemblies in excess of one region shall not be discharged before the interval of 400 hours has elapsed after shutdown.

9. Whenever movement of irradiated fuel is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of the reactor pressure vessel flange.

10. Hoists or cranes utilized in handling irradiated fuel shall be dead-load tested before movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than maximum load to be assumed by the hoists or cranes during the refueling operation. A thorough visual inspection of the hoists or cranes shall be made after the dead-load test and prior to fuel handling. A test of interlocks shall also be performed.

11. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

12. To ensure redundant decay heat removal capability, at least two of the following requirements shall be met:

    a. No. 31 residual heat removal pump and heat exchanger, together with their associated piping and valves is operable.

    b. No. 32 residual heat removal pump and heat exchanger, together with their associated piping and valves is operable.

    c. The water level in the refueling cavity above the top of the reactor vessel flange is equal to or greater than 23 feet.

The requirement for the fuel storage building emergency ventilation system to be operable is established in accordance with standard testing requirements to assure that the system will function to reduce the offsite dose to within acceptable limits in the event of a fuel-handling accident. The system is actuated upon receipt of a signal from the area high activity alarm or by a manually-operated switch. The system is tested prior to fuel handling and is in a standby basis.

When fuel in the reactor is moved before the reactor has been subcritical for at least 365 hours, the limitations on the containment vent and purge system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere.

The limit to have at least two means of decay heat removal operable ensures that a single failure of the operating RHR System will not result in a total loss of decay heat removal capability. With the reactor head removed and 23 feet of water above the vessel flange, a large heat sink is available for core cooling. Thus, in the event of a single component failure, adequate time is provided to initiate diverse methods to cool the core.

The minimum spent fuel pit boron concentration and the 90-day restriction of the movement of the spent fuel cask to allow the irradiated fuel to decay were specified in order to minimize the consequences of an unlikely sideways cask drop.

When the spent fuel cask is being placed in or removed from its position in the spent fuel pit, mechanical stops incorporated in the bridge rails make it impossible for the bridge of the crane to travel further north than a point directly over the spot reserved for the cask in the pit. Thus, it will be possible to handle the spent fuel cask with the 40-ton hook and to move new fuel to the new fuel elevator with a 5-ton hook, but it will be impossible to carry any object over the spent fuel storage area with either the 40 or 5-ton hook of the fuel storage building crane.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

INTRODUCTION

A number of events have occurred at operating Pressurized Water Reactor (PWR) facilities where decay heat removal capability has been seriously degraded due to inadequate administrative controls during shutdown modes of operation. One of these events, described in IE Information Notice 80-20 (Ref. 1), occurred at the Davis-Besse Station, Unit No. 1, on April 19, 1980. In IE Bulletin 80-22 (Ref. 2) dated May 9, 1980, licensees were requested to immediately implement administrative controls which would ensure that proper means are available to provide redundant methods of decay heat removal. While the function of the bulletin was to effect immediate action with regard to this problem, the NRC considered it necessary that an amendment be made to each PWR license to provide for permanent long-term assurance that redundancy in decay heat removal capability will be maintained. By the letter dated June 11, 1980 (Ref. 3), all PWR licensees were requested to propose Technical Specification changes that provide for redundancy in decay heat removal capability in all modes of operation, to use the NRC model Technical Specifications to provide an acceptable solution to the concern, to include an appropriate safety analysis as a basis, and to submit the proposed Technical Specification changes along with the bases by October 11, 1980.

The Power Authority of the State of New York, the licensee, responded to the NRC request for amended Technical Specifications for the Indian Point Nuclear Generating Unit 3 with their letter dated July 6, 1983 (Ref. 4) and proposed revisions to the Technical Specifications for the plant. In a letter dated December 3, 1984 (Ref. 5), the licensee responded to the NRC staff's request for additional information and indicated that the existing and previously proposed Technical Specifications supplemented by the surveillance requirements contained in the operating procedures presently available satisfied the NRC requirements of assuring redundant decay heat removal for all modes of reactor operation.

DISCUSSION AND EVALUATION

The proposed Technical Specifications provide for redundant means of decay heat removal in all modes of operation except during refueling when a large mass of water is above the core. These redundant means are outlined as follows:

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Power Operation and Startup

All reactor coolant loops must be in operation with all four reactor coolant pumps operating at power levels above 10% rated power, and when the reactor is critical and above 2% rated power, at least two RC pumps shall be in operation.

Hot Standby (Hot Shutdown; T_{avg} equal to or greater than 350°F)

All reactor coolant loops are required to be operable; however, only one of the loops is required to be in operation.

Hot Shutdown

At least two of the heat removal coolant loops are required to be operable, i.e., either two RC loops or two DHR loops, or a combination of each one of these coolant loops. One of the loops must be in operation.

Cold Shutdown and Refueling with the Water Level Above the Core Less Than 23 Feet

Two decay heat removal loops are required to be operable with at least one loop in operation.

Refueling with the Water Level Above the Core Greater Than 23 Feet

At least one decay heat removal loop is required to be in operation. The other loop need not be operable.

In addition to the above requirements for operability, the Technical Specification revisions specify surveillance intervals for heat removal systems that are consistent with the Standard Technical Specifications. The revised Technical Specifications provided an improvement over the existing ones since redundant decay heat removal will now be provided in Hot Shutdown and Cold Shutdown Modes. In the refueling Mode, with a large mass of water above the core, only a single heat removal path is required. The surveillance requirements that would identify any inoperable equipment or degraded performance are performed during each shift. The staff therefore concludes that the proposed Technical Specifications meet the intent of the Standard Technical Specifications with respect to providing redundant means of decay removal capability in all operating modes and, based on our review, are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 previously issued a proposed finding that this amendment involves no

significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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: April 8, 1985

PRINCIPAL CONTRIBUTOR:

V. Leung

Reference

1. NRC IE Information Notice 80-20, May 8, 1980.

2. NRC IE Bulletin 80-12, May 9, 1980.

3. NRC letter, D. G. Eisenhut, To All Operating Pressurized Water Reactors (PWRs), dated June 11, 1980.

4. New York Power Authority Letter, J. P. Bayne to S. A. Varga, NRC, July 6, 1983.

5. New York Power Authority Letter, C. A. McNeill to S. A. Varga, NRC, December 3, 1984.

6. NRC Memorandum, R. M. Bernero, to D. G. Eisenhut, "Technical Specification Discrepancies," dated September 27, 1984.