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

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FEBRUARY 2 1 1980

Docket No. 50-286

Am-29 to DPR-64

Mr. George T. Berry, Executive Director
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 29 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letters dated December 3, 1979, December 18, 1979 and January 4, 1980.

The amendment changes the Technical Specifications to increase the operability requirements for the auxiliary feedwater pumps and to limit the control rod misalignment, and adds a license condition to require a secondary water chemistry monitoring program.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosure:
1. Amendment No. 29 to DPR-64
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosure
See next page

February 21, 1980

Mr. George T. Berry
Power Authority of the State of New York

cc: White Plains Public Library
100 Martine Avenue
White Plains, New York 10601

Mr. Vito J. Cassan
Assistant General Counsel
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Anthony Z. Roisman
Natural Resources Defense Council
917 - 15th Street, N.W.
Washington, D. C. 20005

Dr. Lawrence R. Quarles
Apartment 51
Kendal at Longwood
Kennett Square, Pennsylvania 19348

Mr. George M. Wilverding
Licensing Supervisor
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Mr. P. W. Lyon
Manager - Nuclear Operations
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Director, Technical Development
Programs
State Of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Mr. J. P. Bayne, Resident Manager
Indian Point 3 Nuclear Power Plant
P. O. Box 215
Buchanan, New York 10511

Mr. J. W. Blake, Ph.D., Director
Environmental Programs
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Theodore A. Rebelowski
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

Joyce P. Davis, Esquire
Law Department
Consolidated Edison Company of
New York Inc.
4 Irving Place
New York, New York 10003

Jeffrey C. Cohen, Esquire
New York State Energy Office
Swan Street Building
CORE 1 - Second Floor
Empire State Plaza
Albany, New York 12223

Honorable George Begany
Mayor, Village of Buchanan
188 Westchester Avenue
Buchanan, New York 10511

Ms. Ellyn Weiss
Sheldon, Harmon and Weiss
1725 I Street, N.W.
Suite 506
Washington, D. C. 20006

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. 29
License No. DPR-64

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Power Authority of the State of New York (the licensee) dated December 3, 1979, December 18, 1979 and January 4, 1980, comply 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;

   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 by changing paragraph 2.B.(2), renumbering present final paragraph 2.I to 2.J, and adding a new paragraph 2.I to read as follows:

   2.B.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   I. The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

      1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

      2. Identification of the procedures used to quantify parameters that are critical to control points;

      3. Identification of process sampling points;

      4. Procedure for the recording and management of data;

      5. Procedures defining corrective actions for off control point chemistry conditions; and

      6. A procedure indentifying the authority responsible for the interpretation of the data, and the sequence and timing of administrative events required to initiate corrective action.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 21, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 29

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.4-1	3.4-1
3.4-2	3.4-2
3.4-3	3.4-3
3.4-6	3.4-6
3.4-10	3.4-10
3.4-16	3.4-16

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System operation is necessary to ensure the capability to remove decay heat from the core.

Specification

A. The reactor shall not be heated above 350°F unless the following conditions are met:

(1) A minimum ASME Code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Three out of three auxiliary feedwater pumps must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tank.

(4) System piping and valves directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) Two steam generators capable of performing their heat transfer function.

(7) City water system piping and valves directly associated with providing backup supply to the auxiliary feedwater pumps are operable.

B. If during power operations any of the conditions of 3.4-A above, except Item (2), cannot be met within 48 hours, the operator shall start to shutdown and cool the reactor below 350°F using normal operation procedures If Item (2) cannot be met within 72 hours, the reactor shall be in hot shutdown within the next 12 hours.

C. The gross turbine-generator electrical output at all times shall be within the limitation of Figure 3.4-1 or Figure 3.4-2 for the application conditions of turbine overspeed setpoint, number of operable low pressure steam dump lines, and condenser backpressure as noted thereon.

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to feed the steam generators is provided by operation of the turbine cycle feedwater system. The twenty main steam safety valves have a total combined rated capability of 15, 108,000 lbs/hr. The total full power steam flow is 12,974,500 lbs/hr, therefore twenty (20) main steam safety valves will be able to relieve the total steam flow if necessary.

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant. The minimum amount of water in the condensate storage tank is the amount needed for 24 hours at hot shutdown. When the condensate storage supply is exhausted, city water will be used.

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove core decay heat after a reactor shutdown.

The limitations placed on turbine-generator electrical output due to conditions of turbine overspeed setpoint, number of operable steam dump lines, and condenser back pressure are established to assure that turbine overspeed (during conditions of loss of plant load) will be within the design overspeed value considered in the turbine missile analysis.[2] In the preparation of Figures 3.4-1 and 3.4-2, the specified number of operable L.P. steam dump lines is shown as one (1) greater than the minimum number required to act during a plant trip. The limitations on electrical output, as indicated in Figures 3.4-1 and 3.4-2, thus consider the required performance of the L.P. Steam Dump System in the event of a single failure for any given number of operable dump lines.

3.10.5 Rod Misalignment Limitations

3.10.5.1 If a control rod is misaligned from its bank demand position by more than 12 steps (indicated position), then realign the rod or determine the core peaking factors within 2 hours and apply Specification 3.10.2.

3.10.5.2 If the requirements of Specification 3.10.3 are determined not to apply and the core peaking factors have not been determined within two hours and the rod remains misaligned, the high reactor flux setpoint shall be reduced to 85% of its rated value.

3.10.5.3 If the misaligned control rod is not realigned within 8 hours the rod shall be declared inoperable.

3.10.6 Inoperable Rod Position Indicator Channels

3.10.6.1 If a rod position indicator channel is out of service then:

a. For operation between 50 percent and 100 percent of rating, the position of the control rod shall be checked indirectly by core instrumentation (excore detectors and/or movable incore detectors) every shift, or subsequent to rod motion exceeding 24 steps, whichever occurs first.

b. During operation below 50 percent of rating, no special monitoring is requried.

3.10.6.2 Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

(e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design basis including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomolies which would, otherwise, affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 12 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.

2. Control Rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.

3. The control rod bank insertion limits are not violated.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during these tests.

The rod position indicator channel is sufficiently accurate to detect a rod $\pm$7 inches away from its demand position. An indicated misalignment less than 12 steps does not exceed the power peaking factor limits. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 4 week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

The required drop time to dashpot entry is consistent with safety analysis.

<u>REFERENCE</u>

1. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program", August 1975
2. FSAR Appendix 14C

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 29 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

By letters dated December 3, 1979, December 18, 1979 and January 4, 1980, the Power Authority of the State of New York (the licensee) requested amendment of facility operating license DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment would make several changes to the Technical Specifications and impose a license condition. The proposed changes would increase the operability requirements for the auxiliary feedwater pumps and limit control rod misalignment, and modify the license to incorporate a secondary water chemistry monitoring program requirement.

Secondary Water Chemistry

By letter dated August 1, 1979, we requested that requirement for a secondary water chemistry monitoring program to inhibit steam generator tube degradation be incorporated in the body of the license. By letter of December 3, 1979 the licensee proposed a modification to the license that would incorporate the program we requested.

This license condition would provide assurance that the licensee will devote proper attention to controlling secondary water chemistry, while also providing the needed flexibility to allow the licensee to more effectively deal with any off-normal conditions that might arise. Moreover, we have concluded that such a license condition, in conjunction with existing Technical Specifications on steam generator tube leakage and inservice inspection, would provide the most practical and comprehensive means of assuring that steam generator tube integrity is maintained.

Therefore, we find the proposed license condition acceptable.

Auxiliary Feedwater Pumps

By letter dated November 7, 1979 we identified requirements for the auxiliary feedwater (AFW) system. NRC staff recommendation GS-1 called for the licensee to propose a Technical Specification requirement to limit the time that only one AFW system pump and its associated flow train and essential instrumentation can be operable. The proposed Technical Specifications were submitted by letter dated January 4, 1980.

The outage time limit as initially proposed was not acceptable. After discussions, the licensee agreed to the outage time limit of 72 hours which was called for in our recommendation GS-1. We, therefore, conclude that the proposed change to the Technical Specifications, with this change in the outage time limit is acceptable.

Control Rod Misalignment

By letter dated October 29, 1979, we identified a problem with control rod misalignment specification for Westinghouse-designed reactors. The Westinghouse safety analyses are performed for control rod misalignments up to ±15 inches, which is equivalent to ±24 steps. Because the control rod position indication system has an uncertainty of 7.5 inches (12 stepas), there could be an actual misalignment of up to 24 steps for an indicated misalignment of 12 steps.

The proposed change to the Technical Specifications, submitted by letter dated December 18, 1979, would limit the indicated control rod misalignment to no more than 12 steps from its control rod bank demand position. We find these proposed Technical Specifications acceptable.

Pending formal licensing actions the licensee committed to administrative controls to limit the indicated control rod misalignment to ±12 steps.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: February 21, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-286

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. to Facility Operating License No. DPR-64, issued to the Power Authority of the State of New York (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 3 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment changes the Technical Specifications to increase the operability requirements for the auxiliary feedwater pumps and to limit control rod misalignment, and adds a license condition to require a secondary water chemistry monitoring program.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated December 3, 1979, December 18, 1979 and January 4, 1980. (2) Amendment No. 29 to License No. DPR-64, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the White Plains Public Library, 100 Martine Avenue, White ~~Plains~~ Plains, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 21 day of February 1980

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors