Docket No. 50-247

February 22, 1973

Consolidated Edison Company
of New York, Inc.
ATTN: Mr. William E. Caldwell, Jr.
Vice President
4 Irving Place
New York, New York 10003

Change No. 2
License No. DPR-26

Gentlemen:

Your letter dated February 9, 1973, requested a change in the Technical Specifications attached as Appendix A to Facility Operating License No. DPR-26 authorizing fuel loading and subcritical testing for Indian Point Unit No. 2. This change was requested to permit loading of refabricated fuel and resumption of subcritical testing pursuant to Facility Operating License No. DPR-26.

This change involves the modifications recently made to the Indian Point Unit No. 2 fuel core consisting principally of pre-pressurization of fuel rods, increased average uranium dioxide density, increased average enrichment in initial uranium 235, and increased burnable poison rods. The prepressurized rods are very similar in design to those previously reviewed for many power reactor facilities and are acceptable. The average uranium dioxide density and the average enrichment of uranium 235 in the fuel has been increased somewhat and therefore additional burnable poison rods have been included in previously vacant rod cluster control guide tubes. The additional burnable poison rods will maintain a non-positive moderator coefficient at the beginning of life. The minimum boron concentration of the refueling water has been increased to maintain the minimum shutdown margin during refueling.

Your document entitled "Fuel densification - Indian Point Nuclear Generating Station, Unit No. 2," transmitted by letter dated January 18, 1973, provides the revised safety analyses regarding core performance during power operation including normal operation, design transients, and accidents. Additional staff review of this document is required to provide assurance of satisfactory core performance during power operation.

RC

OFFICE ▶	8111060388 730222					
	PDR ADOCK 05000247					
SURNAME ▶	P PDR					
DATE ▶						

However, as Operating License No. DPR-26 authorizes only fuel loading and subcritical testing and no power operation, we are authorizing this Technical Specification Change to DPR-26 as requested.

We have concluded that the proposed change does not involve significant hazard considerations not described or implicit in the Final Facility Description and Safety Analysis Report and that there is reasonable assurance that the health and safety of the public will not be endangered.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specification change outlined in your letter dated February 9, 1973, is hereby authorized. To effect this change, Technical Specification Nos. 3.8 and 5.3 are revised as indicated below:

Technical Specification No. 3.8

1. Basis (p. 3.8-3) in the second paragraph, line 5, change "1564 ppm" to "1615 ppm."

Technical Specification No. 5.3

1. 5.3.A.1 (p. 5.3-1) - Change the second sentence to read: "The pellets are encapsulated in Zircaloy-4 tubing, pre-pressurized with Helium to approximately 450 psig, to form fuel rods."

2. 5.3.A.2 (p. 5.3-1) - In line 1, change "2.70" to "2.76." In line 3, change "3.20" to "3.30."

3. 5.3.A.4 (p. 5.3-2) - In line 2, change "1160" to "1412." In line 2, change "6, 12, and 16-rod" to "7, 8, 9, 12, 16, and 20-rod."

References

1. (p. 3.8-5) - In reference (2), delete "FSAR - Table 3.2.1-1" and add "Fuel Densification - Indian Point Nuclear Generating Station Unit No. 2, dated January 1973, Table 3.3."

2. (p. 5.3-2) - In reference (1), add "Application for a Special Nuclear Material License, dated October 6, 1972, Section III.B."

OFFICE ▶						
SURNAME ▶						
DATE ▶						

(p. 5.3-2) - In reference (2), delete "FSAR Section 3.2.1" and add "Application for a Special Nuclear Material License, dated October 6, 1972, Table 1."

(p. 5.3-2) - In reference (3), delete FSAR Section 3.2.3," and add "Fuel Densification - Indian Point Nuclear Generating Station Unit No. 2, dated January 1973, Figure 3.3."

The Technical Specifications of Facility Operating License No. DPR-26 are hereby changed as set forth in the revised pages 3.8-3, 3.8-5, 5.3-1, and 5.3-2 which are enclosed.

Sincerely,

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosures:
As stated above

cc: Samuel W. Jensch, Esquire
Chairman, Atomic Safety and
Licensing Board
U.S. Atomic Energy Commission
Washington, D. C.

Dr. John C. Geyer, Chairman
Department of Geography and
Environmental Engineering
The John Hopkins University
Baltimore, Maryland 21218

Mr. R. B. Briggs, Director
Molten-Salt Reactor Program
Oak Ridge National Laboratory
P. O. Box Y
Oak Ridge, Tennessee 37830

OFFICE ▶	PWR-2	PWR-2	OGC	AD/PWRs		
7701 SURNAME ▶	MMcCoy:nlg	KKniel	MKarman	RCDeYoung		
DATE ▶	2/22/73	2/22/73	2/22/73	2/22/73		

Consolidated Edison Company
of New York, Inc.

ccs:

Leonard M. Trosten, Esquire
LeBoeuf, Lamb, Leiby & MacRae
1821 Jefferson Place, NW
Washington, D. C. 20036

J. Bruce MacDonald, Esquire
New York State Atomic Energy Council
99 Washington Avenue
Albany, New York 12210

Angus Macbeth, Esquire
Natural Resources Defense Council, Inc.
36 West 44th Street
New York, New York 10036

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, NW
Washington, D. C. 20036

Paul S. Shemin, Esquire
New York State Attorney General's Office
80 Centre Street
New York, New York 10013

Myron Karman, Esquire
Counsel, Regulatory Staff
U.S. Atomic Energy Commission
Washington, D. C. 20545

Honorable William J. Burke
Mayor of the Village
of Buchanan
Buchanan, New York 10511

Atomic Safety and
Licensing Board Panel
U.S. Atomic Energy
Commission
Washington, D. C. 20545

Atomic Safety and
Licensing Appeal
Board
U.S. Atomic Energy
Commission
Washington, D. C. 20545

Mr. Frank W. Karas, Chief
Public Proceedings Branch
U.S. Atomic Energy
Commission
Washington, D. C. 20545

DISTRIBUTION:
AEC PDR
Local PDR
Docket file
RP Reading
PWR-2 Reading
Attorney, OGC
RSBoyd
RO (3)
RCDeYoung
SHanauer, DR
DSkovholt
PCollins

PWR Branch Chiefs
RWKlecker
RVollmer
NDube
MJinks (w/4 enclosures)
MMcCoy
KKniel

bcc: H. Mueller, GMR/H
JRBuchanan, ORNL
TWLaughlin, DTIE
FWKaras, SECY

OFFICE ▶						
SURNAME ▶						
DATE ▶						

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket No. 50-247

February 22, 1973

Consolidated Edison Company
of New York, Inc.
ATTN: Mr. William E. Caldwell, Jr.
Vice President
4 Irving Place
New York, New York 10003

Change No. 2
License No. DPR-26

Gentlemen:

Your letter dated February 9, 1973, requested a change in the Technical Specifications attached as Appendix A to Facility Operating License No. DPR-26 authorizing fuel loading and subcritical testing for Indian Point Unit No. 2. This change was requested to permit loading of refabricated fuel and resumption of subcritical testing pursuant to Facility Operating License No. DPR-26.

This change involves the modifications recently made to the Indian Point Unit No. 2 fuel core consisting principally of pre-pressurization of fuel rods, increased average uranium dioxide density, increased average enrichment in initial uranium 235, and increased burnable poison rods. The prepressurized rods are very similar in design to those previously reviewed for many power reactor facilities and are acceptable. The average uranium dioxide density and the average enrichment of uranium 235 in the fuel has been increased somewhat and therefore additional burnable poison rods have been included in previously vacant rod cluster control guide tubes. The additional burnable poison rods will maintain a non-positive moderator coefficient at the beginning of life. The minimum boron concentration of the refueling water has been increased to maintain the minimum shutdown margin during refueling.

Your document entitled "Fuel densification - Indian Point Nuclear Generating Station, Unit No. 2," transmitted by letter dated January 18, 1973, provides the revised safety analyses regarding core performance during power operation including normal operation, design transients, and accidents. Additional staff review of this document is required to provide assurance of satisfactory core performance during power operation.

However, as Operating License No. DPR-26 authorizes only fuel loading and subcritical testing and no power operation, we are authorizing this Technical Specification Change to DPR-26 as requested.

We have concluded that the proposed change does not involve significant hazard considerations not described or implicit in the Final Facility Description and Safety Analysis Report and that there is reasonable assurance that the health and safety of the public will not be endangered.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specification change outlined in your letter dated February 9, 1973, is hereby authorized. To effect this change, Technical Specification Nos. 3.8 and 5.3 are revised as indicated below:

Technical Specification No. 3.8

1. Basis (p. 3.8-3) in the second paragraph, line 5, change "1564 ppm" to "1615 ppm."

Technical Specification No. 5.3

1. 5.3.A.1 (p. 5.3-1) - Change the second sentence to read: "The pellets are encapsulated in Zircaloy-4 tubing, pre-pressurized with Helium to approximately 450 psig, to form fuel rods."

2. 5.3.A.2 (p. 5.3-1) - In line 1, change "2.70" to "2.76." In line 3, change "3.20" to "3.30."

3. 5.3.A.4 (p. 5.3-2) - In line 2, change "1160" to "1412." In line 2, change "6, 12, and 16-rod" to "7, 8, 9, 12, 16, and 20-rod."

References

1. (p. 3.8-5) - In reference (2), delete "FSAR - Table 3.2.1-1" and add "Fuel Densification - Indian Point Nuclear Generating Station Unit No. 2, dated January 1973, Table 3.3."

2. (p. 5.3-2) - In reference (1), add "Application for a Special Nuclear Material License, dated October 6, 1972, Section III.B."

(p. 5.3-2) - In reference (2), delete "FSAR Section 3.2.1" and add "Application for a Special Nuclear Material License, dated October 6, 1972, Table 1."

(p. 5.3-2) - In reference (3), delete FSAR Section 3.2.3," and add "Fuel Densification - Indian Point Nuclear Generating Station Unit No. 2, dated January 1973, Figure 3.3."

The Technical Specifications of Facility Operating License No. DPR-26 are hereby changed as set forth in the revised pages 3.8-3, 3.8-5, 5.3-1, and 5.3-2 which are enclosed.

Sincerely,

Karl R. Goller /for

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosures:
As stated above

cc: Samuel W. Jensch, Esquire
Chairman, Atomic Safety and
Licensing Board
U.S. Atomic Energy Commission
Washington, D. C.

Dr. John C. Geyer, Chairman
Department of Geography and
Environmental Engineering
The John Hopkins University
Baltimore, Maryland 21218

Mr. R. B. Briggs, Director
Molten-Salt Reactor Program
Oak Ridge National Laboratory
P. O. Box Y
Oak Ridge, Tennessee 37830

ccs:

Leonard M. Trosten, Esquire
LeBoeuf, Lamb, Leiby & MacRae
1821 Jefferson Place, NW
Washington, D. C. 20036

J. Bruce MacDonald, Esquire
New York State Atomic Energy Council
99 Washington Avenue
Albany, New York 12210

Angus Macbeth, Esquire
Natural Resources Defense Council, Inc.
36 West 44th Street
New York, New York 10036

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, NW
Washington, D. C. 20036

Paul S. Shemin, Esquire
New York State Attorney General's Office
80 Centre Street
New York, New York 10013

Myron Karman, Esquire
Counsel, Regulatory Staff
U.S. Atomic Energy Commission
Washington, D. C. 20545

Honorable William J. Burke
Mayor of the Village
of Buchanan
Buchanan, New York 10511

Atomic Safety and
Licensing Board Panel
U.S. Atomic Energy
Commission
Washington, D. C. 20545

Atomic Safety and
Licensing Appeal
Board
U.S. Atomic Energy
Commission
Washington, D. C. 20545

Mr. Frank W. Karas, Chief
Public Proceedings Branch
U.S. Atomic Energy
Commission
Washington, D. C. 20545

B. If any of the specified limiting conditions for refueling is not met, refueling shall cease until the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

Basis

The equipment and general procedures to be utilized during refueling are discussed in the FSAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.(1) Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (2 above) and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The shutdown margin indicated in Part 5 will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 350,000 gallons of water from the refueling water storage tank with a boron concentration of 2000 ppm. The minimum boron concentration of this water at 1615 ppm boron is sufficient to maintain the reactor subcritical by at least 10% $\Delta k/k$ in the cold condition with all rods inserted, and will also maintain the core subcritical even if no control rods were inserted into the reactor.(2) Periodic checks of refueling water boron concentration insure the proper shutdown margin. Part 6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement. 2

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The presence of a licensed senior reactor operator at the site and designated in charge provides qualified supervision of the refueling operation during changes in core geometry.

References

(1) FSAR - Section 9.5.2

(2) Fuel Densification - Indian Point Nuclear Generating Station Unit No. 2, dated January 1973, Table 3.3. 2

5.3 REACTOR

Applicability

Applies to the reactor core, reactor coolant system, and emergency core cooling systems.

Objective

To define those design features which are essential in providing for safe system operations.

A. Reactor Core

1. The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing, prepressurized with Helium to approximately 450 psig, to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.[1] 2

2. The average enrichment of the initial core is a nominal 2.76 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.30 weight per cent of U-235.[2]

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 3.4 weight per cent of U-235.

4. Burnable poison rods are incorporated in the initial core. There are 1412 poison rods in the form of 7, 8, 9, 12, 16, and 20-rod 2
clusters, which are located in vacant rod cluster control guide tubes.[3] The burnable poison rods consist of borated pyrex glass clad with stainless steel.[4]

5. There are 53 full-length RCC assemblies and 8 partial-length RCC assemblies in the reactor core. The full-length RCC assemblies contain a 142-inch length of silver-indium-cadmium alloy clad with the stainless steel. The partial-length RCC assemblies contain a 36-inch length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with Al_2O_3.[5]

B. <u>Reactor Coolant System</u>

1. The design of the reactor coolant system complies with the code requirements.[6]

2. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand the maximum potential seismic ground acceleration, 0.15g, acting in the horizontal and 0.10g acting in the vertical planes simultaneously with no loss of function.

3. The total liquid volume of the reactor coolant system, at rated operating conditions, is 11,350 cubic feet.

<u>References</u>

(1) FSAR Section 3.2.2, Application for A Special Nuclear Materials License dated October 6, 1972, Section IIIB.

(2) Application for a Special Nuclear Materials License, dated October 6, 1972, Table 1. 2

(3) Fuel Densification - Indian Point Nuclear Generating Station Unit No. 2, dated January 1973, Figure 3.3

(4) FSAR Section 3.2.3

(5) FSAR Sections 3.2.1 and 3.2.3

(6) FSAR Table 4.1.9