UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 176
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of of New York, Inc. (the licensee) dated December 10, 1993, and supplemented on August 11, 1994, 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-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 176, 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 prior to startup from the next refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael J. Case

Michael J. Case, Acting 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: September 29, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 176

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	-
5.3-1	5.3-1
5.3-2	5.3-2

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT: REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, and coolant temperature during four-loop and three-loop operation, and reactor coolant flow during four-loop operation.

Objective

To maintain the integrity of the fuel cladding.

Specifications

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1. The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot region of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters: thermal power, reactor coolant temperature and pressure have been related to DNB through correlations which have been developed to predict the DNB flux and location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB thermal design criterion is that the probability of DNB not occurring on the most limiting rod is at least 95 percent (at a 95 percent confidence level) for any Condition I or II event.

In meeting the DNB design criterion, uncertainties in operating parameter, nuclear and thermal parameters, fuel fabrication parameters, and computer codes must be considered. As described in the FSAR, the effects of these uncertainties have been statistically combined with the correlation uncertainty. Design limit DNBR values have been determined that satisfy the DNB design criterion.

Additional DNBR margin is maintained by performing the safety analyses to a higher DNBR limit. This margin between the design and safety analyses limit DNBR values is used to offset known DNBR penalties (e.g., rod bow and transition core) and to provide DNBR margin for operating and design flexibility.

The curves of Figure 2.1-1 show the loci of points of thermal power Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. These curves are based on a peak nuclear hot channel factor as stated in the Core Operating Limits Report (COLR) and a 1.55 cosine axial power shape.

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 core shall contain 193 fuel assemblies. Each fuel assembly shall consist of 204 Zircaloy-4 or ZIRLO clad fuel rods. Limited substitutions of Zircaloy-4, ZIRLO, or stainless steel filler rods for fuel rods, in accordance with NRC approved applications of fuel rod configurations, may be used. Fuel assembly configurations shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by test or cycle-specific reload analyses to comply with all fuel safety design basis. Each fuel rod shall have a nominal active fuel length of 144 inches. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

2. Deleted

3. The enrichment of reload fuel will be no more than 5.0 weight percent U-235 and will be stored in accordance with Technical Specification 5.4.

4. Deleted

5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with stainless steel (1).

B. REACTOR COOLANT SYSTEM

1. The design of the reactor coolant system complies with the code requirements (2). Design values for system temperature and pressure are 650°F and 2485 psig, respectively.

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 nominal liquid volume of the reactor coolant system, at rated operating conditions, and with 0% Steam Generator tube plugging is 11,350 cubic feet.

References

(1) UFSAR Section 3.2

(2) UFSAR Table 4.1-9