ATTACHMENT A

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
Facility Operating License No. DPR-26
November, 1985

50-247

APPLICATION FOR AMENDMENT REVISING TECH. SPECS

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LIST OF FIGURES

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.

C. The following conditions are applicable to the spent fuel pit anytime it contains fuel.

1. Fuel assemblies to be stored in the spent fuel pit are categorized as either Category A, B or C based on burnup and enrichment limits as specified in Figure 3.8-1. The storage of Category A fuel assemblies within the pit is unrestricted. Category B fuel assemblies shall only be loaded into a spent fuel rack cell whose adjacent cells on all four sides either contain non-fuel materials or Category A fuel assemblies. The storage of Category C fuel assemblies within the pit is unrestricted except that they cannot be loaded adjacent to Category B fuel assemblies.

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The shutdown margin indicated in Part 5 will keep the core subcritical, even if all control rods were withdrawn from the core.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 300,000 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 cold shutdown 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 ensure 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.

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 then one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The 131 hour decay time following plant shutdown and the 23 feet of water above the top of the reactor vessel flanges are consistent with the assumptions used in the dose calculations for fuel-handling accidents both inside and outside of the containment. The analysis of the fuel handling accident inside of the containment is based on an atmospheric dispersion factor ($^{X}/Q$) of 5.1×10^{-4} sec/m^3 and takes no credit for removal of radioactive iodine by charcoal filters. The requirement for the spent fuel storage building charcoal filtration system to be operating when spent fuel movement is being made provides added assurance that the offsite doses will be within acceptable limits in the event of a fuel-handling accident. The additional month of spent fuel decay time will provide the same assurance that the offsite doses are within acceptable limits and therefore the charcoal filtration system would not be required to be operating.

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.

The fuel enrichment and burnup limits in Specification 3.8.C.1 assures the limits assumed in the spent fuel safety analyses will not be exceeded. Within this specification adjacent location means those four locations directly contacting the four sides (faces) of a fuel assembly but excludes those four locations which contact the four corners of a fuel assembly.

References

(1) FSAR - Section 9.5.2

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

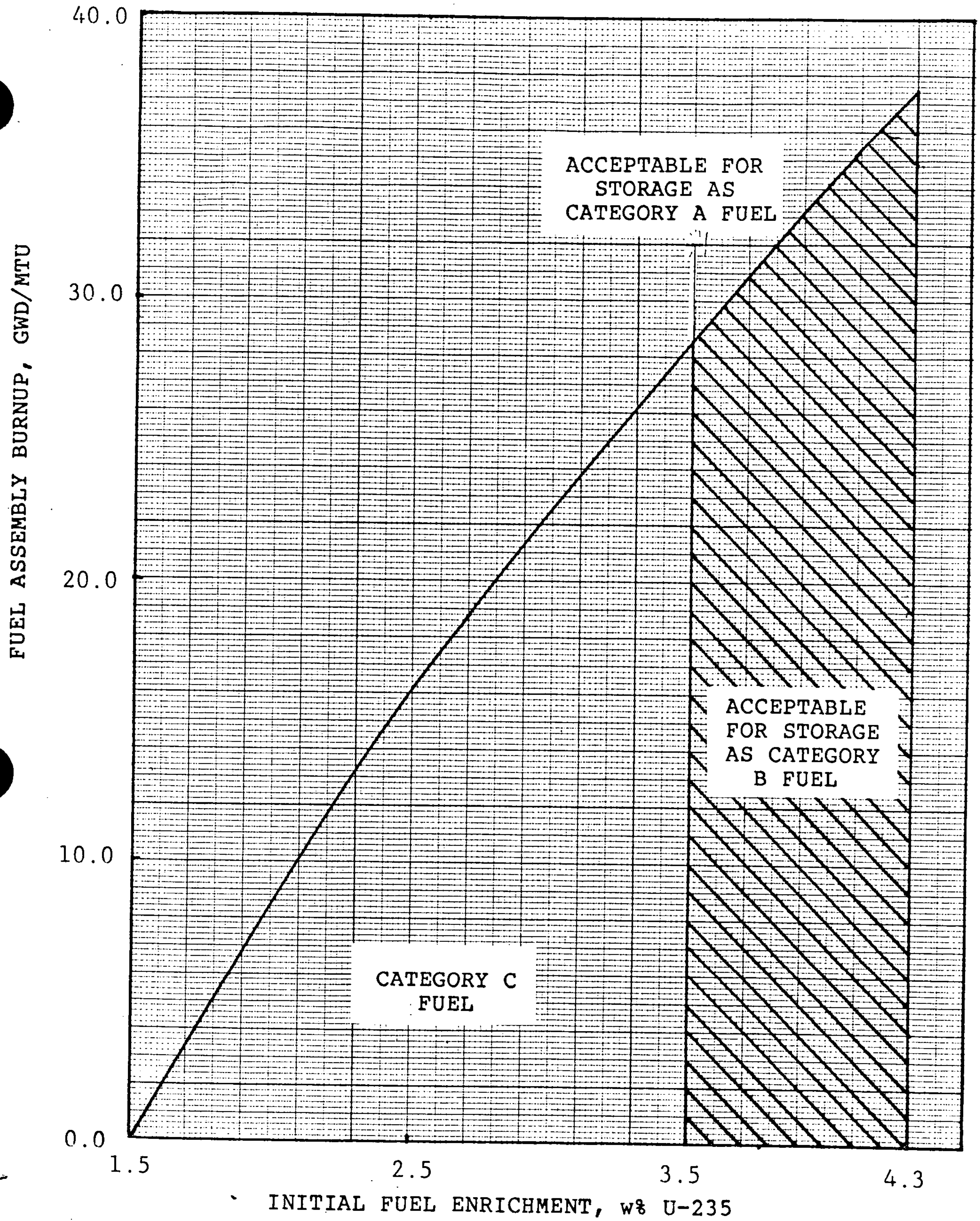

FIGURE 3.8-1

LIMITING FUEL BURNUP VERSUS INITIAL ENRICHMENT

CATEGORY A - AREA ALONG THE CURVE AND ABOVE
CATEGORY B - AREA BELOW THE CURVE AND 3.5 < w% U-235 ≤ 4.3
CATEGORY C - AREA BELOW THE CURVE AND w% U-235 ≤ 3.5

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 to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.(1)

2. Deleted

3. The enrichment of reload fuel will be no more than 4.3 weight per cent 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 the stainless steel.(5)

B. Reactor Coolant System

1. The design of the reactor coolant system complies with the code requirements.(6) Design values for system temperature and pressure are 650^{o}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.

5.4 FUEL STORAGE

Applicability

Applies to the capacity and storage arrays of new and spent fuel.

Objective

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

Specification

1. The spent fuel pit structure is designed to withstand the anticipated earthquake loadings as a Class I structure. The spent fuel pit has a stainless steel liner to insure against loss of water.

2.A. The new fuel storage rack is designed so that it is impossible to insert assemblies in other than an array of vertical fuel assemblies with the sufficient center-to-center distance between assemblies to assure $K_{eff} \leq 0.95$ even if unborated water were used to fill the pit and with the fuel loading in the assemblies limited to 54.33 grams of U-235 per axial centimeter of fuel assembly.

2.B. The spent fuel storage racks are designed and their loading maintained within the limits of Technical Specification 3.8.C.1, such that $K_{eff} \leq 0.95$ even if unborated water were used to fill the pit and with the fuel loading in the assemblies limited to 54.33 grams U-235 per axial centimeter of fuel assembly.

3. Whenever there is fuel in the pit, the spent fuel storage pit is filled and borated to the concentration to match that used in the reactor cavity and refueling canal during refueling operations.

ATTACHMENT B

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Safety Assessment

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
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Safety Assessment

The proposed revisions contained in Attachment A would modify the Indian Point Unit No. 2 Technical Specifications to permit the use of higher enrichment reload fuel assemblies and storage of such assemblies both prior and subsequent to their loading in the reactor. The modification of these specifications is requested in order to facilitate the use of enriched fuel assemblies and to provide the capability for extended fuel cycles. Extended fuel cycles will result in reduced fuel handling which in turn will provide ALARA savings. It should be noted that there are no hardware changes necessary for the proposed increase in reload fuel enrichment.

The increase in reload fuel enrichment from 3.5 w/o U-235 to 4.3 w/o U-235 will enable Con Edison to decrease the required number of fuel assemblies per reload. This will decrease the number of spent fuel assemblies discharged per reload, thereby reducing the required spent fuel pool capacity as compared to present reloads. Supplemental analysis has been performed and is presented in Enclosure 1. The analysis in Enclosure 1 supplements our analysis transmitted by letter dated May 6, 1980 to H.R. Denton from W.J. Cahill, Jr. The analysis in Enclosure 1 demonstrates that the existing spent fuel storage racks at Indian Point 2 can safely store fuel with initial enrichments of 4.3 weight percent of U-235. The results of the analysis show that, with the fuel loading specifications assumed, the criticality design criterion of Keff $\leq$ 0.95 is met. These fuel loading specifications have been incorporated in the proposed technical specification changes. The proposed increase in fuel enrichment may result in higher burnup fuel being stored in the spent fuel pool. The existing spent fuel decay heat load analysis remains unaffected by the potential for higher burnup fuel. In addition, there is no significant impact on offsite dose due to this proposed increase in fuel enrichment.

Analysis has shown that the existing storage rack for new fuel is sufficient to maintain an array of vertical fuel assemblies with enrichments of 4.3 weight percent of U-235, when fully flooded with potential moderators such as unborated water, in a subcritical condition, i.e., Keff $\leq$ 0.95. In addition, analysis has shown that assuming optimum moderating conditions, the existing fuel storage rack for new fuel assemblies is sufficient to assure Keff will not exceed 0.95 with fuel stored with enrichments of 4.3 weight percent of U-235. Enclosure 2 to this application is a summary of the criticality analysis performed for the new fuel assembly storage rack including the identification of the computer codes used.

Technical Specification 5.3.A.3 is proposed to be revised to indicate that the reload fuel enrichment will be no more then 4.3 weight percent of U-235. Technical Specification 5.4.2 is proposed to be revised to indicate the fuel loading limits, corresponding to 4.3 weight percent of U-235, for the fuel racks for both new and spent assemblies. Technical Specification 3.8.C.1 is proposed to be included in order to incorporate the restrictions assumed in the analysis of Enclosure 1.

Basis For No Significant Hazards Consideration Determination

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazard consideration exists by providing certain examples (48 FR 14870). Example (vi) of those involving no significant hazards considerations discusses a change which may reduce a safety margin but where the results are clearly within all acceptable criteria with respect to the system or component. The proposed increase in reload fuel enrichment and the fuel rack loading limits for both new and spent assemblies is in a less restrictive direction and would appear to reduce a safety margin. However, the proposed change is based on conservative analyses which show that, with the fuel loading specified in proposed technical specification 3.8.C.1 for fuel racks for spent assemblies and 5.4.2.A for the fuel rack for new assemblies, the criticality design criteria of Keff $\leq$ 0.95 is met. Therefore, consistent with the Commission's criteria for determining a proposed amendment to an operating license involves no significant hazard considerations, 10 CFR 50.92 (48 FR 871), we have determined that the proposed change to increase the new and spent fuel rack loading limits will not increase the probability or the consequences of an accident previously evaluated, or create the possibility of a new or different kind of accident from any previously evaluated, or involve a significant reduction in a margin of safety.

Therefore, since this application for amendment involves a proposed change that is similar to an example for which no significant hazards consideration exists, we have determined that this application involves no significant hazards consideration.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both committees concur that these changes do not represent a significant hazards consideration and will not cause any change in the types or increase in the amounts of effluents or any change in the authorized power level of the facility.