November [illegible], 1989

MEMORANDUM FOR: Don Brinkman, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II

FROM: LeMoine J. Cunningham, Chief
Radiation Protection Branch
Division of Radiation Protection
and Emergency Preparedness

SUBJECT: SAFETY EVALUATION REPORT AND ENVIRONMENTAL ASSESSMENT FOR INDIAN POINT UNIT NO. 2 SPENT FUEL POOL RERACK (TAC NO. 72962)

The Radiation Protection Branch (PRPB) has reviewed the June 20, 1989 submittal of the Consolidated Edison Company at New York (ConEd) requesting an amendment to License DPR-26 and the revision of Technical Specifications related to the proposed spent fuel pool rerack modification at the Indian Point Unit 2 Nuclear Power Station, and the additional information contained in the October 23, 1989 response to NRR staff questions relevant to this application. Our evaluation includes information on occupational radiation exposure, radioactive wastes, accident analyses, potential releases of radioactive materials, and offsite radiological impacts due to the proposed spent fuel pool rerack.

The radiation protection measures proposed by ConEd for the Indian Point Unit 2 Nuclear Power Station spent fuel pool rerack modification provide adequate assurance that appropriate radiation protection measures will be applied during the tasks and that doses to workers and the general public will be kept as low as is reasonably achievable, and are therefore acceptable. A Safety Evaluation Report (Enclosure 1) and Environmental Assessment (Enclosure 2) summarizing our review are enclosed. Also enclosed is a SALP input based on our review.

This review was conducted by J. L. Minns, (492-3151) and R. Pedersen (492-1079).

Original signed by LeMoine J. Cunningham

LeMoine J. Cunningham, Chief
Radiation Protection Branch
Division of Radiation Protection
and Emergency Preparedness

Enclosures:
1. SER
2. EA
3. SALP

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ENCLOSURE 1

SAFETY EVALUATION REPORT
INDIAN POINT UNIT NO. 2
SPENT FUEL POOL RERACK

The Radiation Protection Branch has reviewed Consolidated Edison Companies (ConEd), plan to modify the spent fuel pool (SFP) at their Indian Point Unit 2 Nuclear Power Station. This plan as described in the technical report enclosed in their June 20, 1989 submittal, and supplemented by their letter dated September 23, 1989, would increase the SFP storage capacity at Indian Point Unit 2 (IP2) from 980 to 1376 assemblies by reracking the pool with new high density racks. The additional occupational radiation exposure for reracking the pool is estimated by the licensee to be less than 10 person-rem. Operation of the pool with an increased storage capacity will not contribute any significant increase in plant occupational exposure.

The existing SFP racks consist of twelve independent free standing modules and can be lifted out of the pool without the use of divers. Similarly the new high density racks are also independent free standing modules and will not need divers to install them in the SFP. In the event that divers are required during this reracking operation, the licensee has specific station procedures in place to insure the radiation exposure received by the divers is ALARA. Each diver will be equipped with a remote-readout radiation detector which will be continuously monitored by a health physics technician. Also each diver will have a calibrated alarming dosimeter. Spent fuel will be relocated to minimize radiation exposure to divers. Radiation surveys in the pool will be performed daily (prior to diving) and whenever fuel is moved. Also QA personnel will independently witness and verify the locations of fuel assemblies whenever fuel is moved.

Based on our review of the IP2 Report, we conclude that the projected activities and estimated person-rem doses for this project appear reasonable. ConEd intends to take ALARA considerations into account, and to implement reasonable dose-reducing activities. We conclude that ConEd will be able to maintain individual occupational radiation exposures within the applicable limits of 10 CFR Part 20, and maintain doses ALARA, consistent with the guidelines of Regulatory Guide 8.8. Therefore, the proposed radiation protection aspect of the SFP rerack is acceptable.

DESIGN BASIS ACCIDENTS

In its application, the licensee evaluated the possible consequences of postulated accidents and included means for their avoidance in the design and operation of the facility, and has provided means for mitigation of their consequences should they occur. The staff independently assessed such so-called design basis accidents (DBAs) and agrees with the licensee that no previously unconsidered DBA would be created by the installation and operation of the reracked spent fuel storage pool.

In its previous Safety Evaluation Report (memo from D. R. Mueller to G. Lainas, "Re-evaluation of Fuel Handling Accident Inside Containment - Indian Point 2" (TAC NO. 51920), dated December 27, 1983,) the staff conservatively estimated offsite doses due to exposures to radionuclides released to the atmosphere from a fuel handling accident. This is the staff's scoping DBA for the spent fuel storage pool. The staff concluded that the plant mitigative features would reduce the DBA doses to well below the doses specified in the applicable regulation at 10 CFR Part 100.

Since the applicant intends to utilize higher enrichment fuel, for which higher burnups are intended, the staff reanalyzed the fuel handling DBA for this case. Increased burnup could increase offsite doses from the fuel handling DBA by a factor of 1.2 (NUREG/CR-5009, February 1988). Burnup to 60,000 MWD/T would require the use of fuel initially enriched to about 5.3 weight percent U-235. Thus, we conservatively increased the previously estimated doses by a factor of 1.2. In Table 1.0, the new and old DBA doses are presented and compared to the guideline doses in 10 CFR Part 100. As shown in this table, the DBA doses are still well within the regulatory guideline values and are, therefore, acceptable.

TABLE 1.0

Radiological Consequences of Fuel
Handling Design Basis Accident

	Exclusion Area	
	Thyroid	Whole Body
Revised Estimates (SER - 1983)	100	0.4
Estimates for Higher Fuel Burnup*	120	0.48
Regulatory Requirement (10 CFR Part 100)	300	25

*Factor of 1.2 greater than original estimate

ENCLOSURE 2

ENVIRONMENTAL ASSESSMENT REGARDING THE PROPOSED SPENT FUEL POOL MODIFICATION OF THE INDIAN POINT 2 NUCLEAR POWER PLANT

1.0 INTRODUCTION

In a letter dated June 6, 1989, the licensee, Consolidated Edison Company of New York, submitted a request for a license amendment for spent fuel pool (SFP) modifications to replace the existing spent fuel storage racks at the Indian Point Unit 2 (IP-2) with maximum density storage racks. This replacement will result in an increase in the spent fuel storage capability of the spent fuel pool. IP-2 currently has a licensed spent fuel storage capacity of 980 spaces. The proposed reracking will increase the capacity to 1376 storage spaces.

2.0 RADIOACTIVE WASTES

The plant contains radioactive waste treatment systems designed to collect and process the gaseous, liquid, and solid waste that might contain radioactive material. The radioactive waste treatment systems are evaluated in the Final Environmental Statement (FES). There will be no change in the waste treatment systems described in the FES because of the proposed SFP replacement.

2.1 Radioactive Material Released to the Atmosphere

The station Radiological Effluent Technical Specifications limit the total releases of gaseous activity and require that releases be continuously monitored to assure that releases are within the regulatory limits of 10 CFR Part 20.

With respect to releases of gaseous materials to the atmosphere, the only radioactive gas of significance which could be attributable to storing additional spent fuel assemblies for a longer period of time would be the noble gas radionuclide Krypton-85 (Kr-85). To determine the average annual release of Kr-85, we assume that all of the Kr-85 released from any defective fuel discharged to the SFP will be released prior to the next refueling. Thus, enlarging the storage capacity of the SFP has no effect on the calculated average annual quantities of Kr-85 released to the atmosphere each year. There may be some small change in the calculated quantities due to a change in the fuel burnup; this is expected to be a small fraction of the calculated annual quantities (also see Section 4.0, below).

Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings.

Most of the tritium in the SFP water results from activation of boron and lithium in the primary coolant. Thus, the tritium concentration from this source will not be affected by the proposed changes. A relatively small amount of tritium is created during reactor operation by fissioning of reactor fuel and subsequent diffusion of tritium through the fuel and fuel cladding.

Tritium release from the fuel essentially occurs while the fuel is hot, that is, during operations and, to a limited extent, shortly after shutdown. Thus, expanding the SFP capacity will not significantly increase the tritium activity in the SFP.

2.2 Solid Radioactive Wastes

The concentration of radionuclides in the pool water is controlled by the SFP cleanup system and by decay of short-lived radionuclides. The activity is highest during refueling operations when reactor coolant water is introduced into the pool, and decreases as the pool water is processed through the SFP cleanup system. The increase of radioactivity, if any, due to the proposed modification, should be minor because of the capability of the cleanup system to continuously remove radioactivity from the SFP water and lower radioactivity to acceptable levels.

We do not expect any significant increase in the amount of solid waste generated from the SFP cleanup systems due to the proposed modification. The expected increase in total waste volume shipped from IP-2 is less than 1 percent and would not have any significant additional environmental impact.

If the present spent fuel racks to be removed from the SFP of IP-2 are contaminated because of the proposed modification, they may be disposed of as low level solid waste. Averaged over the lifetime of the station, this would increase the total waste volume shipped from the station by less than 1 percent. This will not have any significant additional environmental impact.

2.3 Radioactive Material Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modifications.

Since the SFP cooling and cleanup systems operate as a closed system, only water originating from cleanup of SFP floors and resin sluice water need be considered as potential sources of radioactivity. It is expected that neither the flow rate nor the radionuclide concentration of the floor cleanup water will change as a result of these modifications.

The SFP demineralizer resin removes soluble radioactive materials from the SFP water. These resins are periodically sluiced with water to the spent resin storage tank. The amount of radioactivity on the SFP demineralizer resin may increase slightly due to the additional spent fuel in the pool, but the soluble radioactive material should be retained on the resins. Radioactive material that might be transferred from the spent resin to the sluice water will be effectively removed by the liquid radwaste system. After processing in the liquid radwaste system, the amount of radioactivity released to the environment as a result of the proposed modification would be negligible.

2.4 Evaluation of Radiation Doses to Members of the Public

Sections 2.1 and 2.3 indicated that releases to the atmosphere and receiving waters, respectively, would not be significant and would be well within regulatory limits. Consequently, the estimated increase in doses due to exposure of individuals and the population to radioactive materials associated with the spent fuel pool modification will not be significant, i.e., also, well within regulatory limits.

3.0 ASSESSMENT OF OCCUPATIONAL RADIATION EXPOSURE

This section contains the staff's evaluation of the estimates of the additional radiological impacts on the plant workers from the proposed operation of the modified SFP.

The occupational exposure for the proposed modification of the SFP is estimated by the licensee to be less than 10 person-rems. This dose is less than 3 percent of the average annual occupational dose of 410 person-rems per unit per year for operating PWRs in the United States. The small increase in radiation dose should not affect the licensee's ability to maintain individual occupational doses with the limits of 10 CFR 20, and as low as is reasonably achievable. Normal radiation control procedures (NUREG-0800, US NRC 1981) and Regulatory Guide 8.8 (US NRC 1978) should preclude any significant occupational radiation exposures.

Based on present and projected operations in the SFP area, we estimate that the proposed operation of the modified SFP should add only a small fraction to the total annual occupational radiation dose at this facility.

Thus, we conclude that the proposed storage of spent fuel in the modified SFP will not result in any significant increase in doses received by workers.

4.0 RADIOLOGICAL CONSEQUENCE OF POTENTIAL ACCIDENTAL RELEASES

No onsite fuel handling accidents having significant offsite radiological consequence have ever occurred. Such accidents and their potential environmental consequences must be postulated. Potential environmental consequences of postulated accidents may be bounded realistically by extrapolation of results from conservative estimates. Offsite doses are estimated conservatively in NRC staff safety reviews for plant siting, design and operations evaluations. The combination of assumptions used for the conservative dose estimates assure that doses for such design basis accidents (DBAs) are unrealistically high. This helps to assure safe plant siting, design and operations because the doses so calculated would exceed regulatory limits without the adoption of plant safety features and/or operational controls. The principal regulatory dose limits for safety reviews are embodied in the NRC Regulations at 10 CFR Part 100. For safety reviews, the limiting dose is 300 rems to the thyroid, principally due to inhalation of I-131 postulated to be accidentally released to the atmosphere.

Several bounding accident analyses for this current assessment have been reported previously (NUREG-0712), and the potential consequences have been found acceptable by the NRC staff. The only pertinent credible accident that has not been analyzed for this assessment is the postulated damage of fuel being handled during the reracking period, with a concomitant release of radioactivity to the atmosphere. A postulated design basis fuel handling accident has been analyzed previously in the Safety Evaluation Report, and a thyroid dose of 100 rems for a person at the site boundary was conservatively estimated. For purposes here, it is significant that this very conservative estimate was based on postulated damage to fuel which had decayed for only three days. In the present case, however, irradiated fuel will have decayed a minimum of 60 days (this will be assured by licensee-imposed administrative controls). I-131 has a half-life of about eight days. During the additional 60 days, I-131 will decay by an additional factor of about 175. The postulated dose will decrease proportionately.

Thus, regardless of the accident probability, which experience says is very low, the offsite thyroid dose due to this bounding postulated accident can be conservatively estimated as 100/175 = 0.5 rem. This dose would be well below the U. S. Environmental Protection Agency Protective Action Guide of 5 rems (thyroid), for which offsite protective action would be warranted. Thus, based on this bounding analysis, the potential environmental consequences of possible accidents are acceptably low, as are the risks.

5.0 CONCLUSIONS

Based on its review of the proposed expansion of the SFP at IP-2, the staff concludes that:

1. The estimated additional radiation doses to the general public are:

    a. Well within those presented in the staff's Final Environmental Statement relative to the operation of Indian Point Unit 2.

    b. Very small in comparison to the dose members of the public receive each year from exposure to natural background radiation.

2. The licensee has taken appropriate steps to ensure that occupational dose will be maintained as low as is reasonably achievable (ALARA) and within the limits of 10 CFR Part 20. The total occupational dose estimated to be associated with the proposed modification of the expanded fuel pool is less than 10 person-rems, which is a small fraction of the average annual total occupational dose at Indian Point Unit 2.

3. The risks of accidents are very low.

On the basis of the foregoing evaluation, it is concluded that there would be no significant additional environmental radiological impact attributable to the proposed reracking and modification to increase the spent fuel storage capacity at Unit 2 of the Indian Point Nuclear Generating Plant.

We have concluded, based on the considerations discussed above, that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, with regard to radiation doses to the public and plant workers.

6.0 REFERENCES

U.S. Environmental Protection Agency, 1972, ORP-SID-72-1, "Natural Radiation Exposure in the United States," June 1972.

U.S. Nuclear Regulatory Commission, 1973, "Final Environmental Statement Related to Indian Point Nuclear Generating Plant Unit No. 2, September 1975.

USNRC, 1977, Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," October 1977.

USNRC, 1978, Regulatory Guide 8.8, Revision 3, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Achievable," June 1978.

USNRC, 1981, NUREG-0800, "Radiation Protection," in: "Standard Review Plan," Chapter 12, July 1981 (formerly issued as NUREG-75/087).

USNRC, 1985, NUREG-0713, Volume 7, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1985," April 1988.

Memorandum from D. R. Mueller to G. Lainas, "Re-evaluation of Fuel Handling Accident Inside Containment - Indian Point 2" (TAC. 51970), dated December 17, 1983.

ENCLOSURE 3

SALP INPUT

Facility Name: Indian Point Unit 2

SUMMARY OF REVIEW ACTIVITIES:

During this period the Radiation Protection Branch reviewed the radiological controls associated with the proposed reracking of their spent fuel pool.

Resolution of the issues involved one request for additional information from the licensee.

NARRATIVE DISCUSSION OF LICENSEE PERFORMANCE FUNCTIONAL AREA - RADIOLOGICAL CONTROLS:

The licensee's submittal for approval of this action was technically sound and demonstrated a knowledge and responsiveness to NRC staff concerns. The licensee's response to the request for additional information was complete allowing the staff to complete its review without further dialog.