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May 13, 1977

Docket No. 50-247

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 30 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment responds to your application dated May 6, 1977, and provides additional requirements for inspection of the steam generators and for steam generator leakage limits as discussed with your staff.

You are requested to submit the details of the next steam generator inspection which you plan for Indian Point Unit No. 2. These details should be submitted no later than 30 days prior to the date you expect the next inspection to commence.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Signed

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 30
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures: See next page

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*SEE PREVIOUS YELLOW FOR CONCURRENCES

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Sincerely,

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Enclosures:
1. Amendment No.
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures: See next page

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DATE ➤	5/ /77	5/13/77	5/13/77	5/13/77	5/ /77	

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

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. 30
License No. DPR-26

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

   A. The facility will operate in conformity with the provisions of the Atomic Energy Act, of 1954, as amended, and the rules and regulations of the Commission;

   B. 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;

   C. 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

   D. 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 paragraphs 2.C.(2) and 2.D. of Facility Operating License No. DPR-26 are hereby amended to read as follows:

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2.C.(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 30 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

2.D. Steam Generator Inspections

In order to perform an inspection of the steam generators, the plant shall be brought to the cold shutdown condition within twelve equivalent months of operation from May 13, 1977. For the purpose of this requirement, equivalent operation is defined as operation with a primary coolant temperature greater than 350°F. Nuclear Regulatory Commission approval shall be obtained before resuming power operation following this inspection.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 13, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 30

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1-17	3.1-17 & 3.1-17a
3.1-19 & 3.1-20	3.1-19 & 3.1-20

Changed areas on the revised pages are shown by marginal lines. Page 3.1-19 is unchanged and is included for convenience only.

F. LEAKAGE OF REACTOR COOLANT

Specification

1. If leakage of reactor coolant is indicated by the means available such as water inventory balance, monitoring equipment or direct observation, a follow up evaluation of the safety implications shall be initiated as soon as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that either (1) a safety problem does not exist or (2) that the indicated leak cannot be substantiated by direct observation or other indication.

2. If the indicated leakage is substantiated and is evaluated as unsafe or is determined to exceed 10 GPM, reactor shutdown shall be initiated as soon as practicable but no later than within 24 hours after the leak was first detected.

3. Whenever the reactor is shutdown, or a steam generator removed from service, in order to investigate steam generator tube leakage and/or to plug or otherwise repair a leaking tube, Con Ed shall inform the NRC before any tube is plugged or, if no tube is plugged, before the steam generator is returned to service.

4. Primary to secondary leakage through the steam generator tubes shall be limited to 1.0 gpm total. With any steam generator tube leakage greater than this limit the reactor shall be brought to the cold shutdown condition within 24 hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

5. If leakage from two or more tubes in the steam generators in any 20-day period is observed or determined, the reactor shall be brought to the cold shutdown condition within 24 hours and

Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation or if two steam generator tube leaks are observed after the reactor is in cold shutdown Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

6. The nature of the leak as well as the magnitude of the leak shall be considered in the safety evaluation. If plant shutdown is necessary per specification 2 above, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable. The safety evaluation shall assure that the exposure to offsite personnel to radiation from the primary system coolant activity is within the guidelines of 10 CFR 20. The reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

7. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different principles shall be in operation, with one of the two systems sensitive to radioactivity. The system sensitive to radioactivity may be out-of-service for 48 hours provided two other systems are available.

is sensitive are 0.025 gpm to greater than 10 gpm, assuming corrosion product activity and no fuel cladding leakage. Under these conditions, an increase in coolant leakage of 1 gpm is detectable within 1 minute after it occurs.

b. The containment radiogas monitor is less sensitive and is used as a backup to the air particulate monitor. The sensitivity range of the instrument is 10^{-3} μc/cc to 10^{-6} μc/cc. Assuming a constant background radioactivity in the containment atmosphere due to normal leakage of reactor coolant with equilibrium fission product gaseous activity, a 1 gpm coolant leak would double the background in about two hours time.

c. The humidity detectors. This method provides a backup to a. and b. This instrument will be sensitive to incremental increases of water leakage to the containment atmosphere on the order of 0.25 gpm per F degree of dewpoint temperature increase.

d. A leakage detection system which determines leakage losses from all water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary. Condensate flows from approximately 1 gpm to 30 gpm per detector can be measured by this system.

Leaks less than 1 gpm may be determined by periodic observation of the water accumulation in the standpipes of the condensate collection system.

As described above, the four reactor coolant leak detection systems are based on 3 different principles, i.e. activity, humidity and condensate flow measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the containment.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory balances.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by limitation of steam generator leakage between the reactor coolant system and the secondary coolant system. Leakage in excess of 1 gpm total for all four steam generators will require plant shutdown and the leaking tubes will be located and plugged.

References

FSAR Sections 11.2.3 and 14.2.4

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

INTRODUCTION

On December 8, 1976, the Nuclear Regulatory Commission (NRC) issued Amendment No. 23 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2 which required that the unit be brought to cold shutdown condition to inspect the steam generators within six (6) equivalent months of operation from the date of the amendment. In addition, the licensee was also required to submit the details of the steam generator inspection program no later than 30 days prior to the date that the inspection was to commence.

By letter dated February 22, 1977, Consolidated Edison Company (the licensee) submitted details of the inspection program for the Unit 2 steam generators. On March 22, 1977, NRC informed the licensee of its approval of their inspection program with additional requirements which include the gauging of the dented tubes. Subsequently, on April 10, 1977, the unit was brought to the cold shutdown condition to perform the inspection in accordance with the requirements delineated in February 22, 1977, submittal and March 22, 1977, letter from NRC. By letter dated May 6, 1977, Consolidated Edison informed the NRC of the results of their inspection of steam generator tubes and requested NRC concurrence to return the reactor to power operation.

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BACKGROUND

Water Chemistry

For many years a sodium phosphate treatment for PWR secondary coolant was widely used for U-tube design steam generators that removed precipitated or suspended solids by blowdown. It was successful as a scale inhibitor, however, in the early use, many PWR U-tubed steam generators with Inconel-600 tubing experienced stress corrosion cracking. The cracking was attributed to free caustic which can be formed when the Na/PO_4 ratio exceeds the recommended limit of 2.6. In addition, some of the insoluble metallic phosphates, formed by the reaction of sodium phosphates with the dissolved solids in the feedwater, were not adequately removed by blowdown. These precipitated phosphates tended to accumulate as sludge on the tube sheet and tube supports at the central portion of the tube bundle where restricted water flow and high heat flux occurs. Phosphate concentration (hideout) at crevices in areas of the steam generator, noted above, caused localized wastage resulting in thinning of the tube wall. The problem of stress corrosion cracking was corrected by maintaining the Na/PO_4 ratio between 2.6 and 2.3. Although the recommended Na/PO_4 ratio was maintained, it did not correct the phosphate hideout problem that caused wastage of the Inconel-600. Largely to correct the wastage and caustic stress corrosion cracking encountered with the phosphate treatment, most PWRs with a U-tube design steam generator using a phosphate treatment for the secondary coolant have now converted to an all volatile chemistry (AVT).

In 1975, radial deformation, or the so-called "denting," of steam generator tubes occurred in several PWR facilities after 4 to 14 months operation, following the conversion from a sodium phosphate treatment to an AVT chemistry for the steam generator secondary coolant. Tube denting occurs predominantly in rigid regions or so-called "hard spots" in the tube support plates. These hard spots

are located in the tube lanes between the six rectangular flow slots in the support plates near the center of the tube bundle and around the peripherial locations of the support plate where the plate is wedged to the wrapper and shell. The hard spot areas do not contain the array of water circulation holes found elsewhere in the support plates.

The phenomenon of denting has been attributed to the accelerated corrosion of the carbon steel support plates at the tube/tube support plate intersection (annuli). The corrosion product (magnetite) from the carbon steel plate has expanded volumetrically to exert sufficient compressive forces to dent the tube and crack the tube support plate ligaments between the tube holes and water circulation holes, due to an in-plane expansion of the support plate. As a result of the tube support plate expansion, the rectangular flow slots began to "hourglass;" i.e., the central portion of the parallel flow slot walls have moved closer so that some of flow slots are now narrower in the center than at the ends.

U-Bend Cracks

On September 15, 1976, during normal operation, one U-tube in the innermost row parallel to the rectangular flow slots in steam generator A at Surry Unit No. 2 rapidly developed a substantial primary to secondary leak (about 80 gpm). After removal of the damaged tube and subsequent laboratory analysis, it was established that the leak resulted from an axial crack, approximately 4-1/4 inches in length, in the U-bend apex due to intergranular stress corrosion cracking that initiated from the primary side. Since the initial parallel flow slot wall in the top support plate has moved closer, the support plate material around the tubes nearest this central portion of these flow slots has also moved inward, in turn forcing an inward displacement of the legs of the U-bends at these locations. This inward movement of the legs of the U-bends at these locations caused an

increase in the hoop strain and ovality of the tubes at the U-bend apex. It is this additional increase in strain at the apex of the U-bend which is believed to be required to initiate stress corrosion cracking of the Inconel 600 alloy tubing exposed to PWR primary coolant.

Laboratory examination of 71 U-bends removed from flow slot locations in rows 1, 2, and 3 of the Surry Units No. 1 and 2 and Turkey Point Unit No. 4 steam generators has shown that intergranular cracking at the U-bend apex was found only in the row 1 tubes.

Of the 71 tubes removed from these operating reactors, which are the most severely affected, no cracks have been found in tubes with computed equivalent strains less than 13.5% after approximately 11,035 hours of effective full power operation since detection of the first tube dent. However, this same equivalent operating time led to the tube failure at Surry Unit No. 2, where the equivalent strain was estimated to be >14.3%. This indicates a strain level at which rapid development of stress corrosion cracking may occur in U-bends of steam generators of this design.

Recent test work also indicates that long incubation periods are needed for the development of stress corrosion cracking at some strain rates.[2-5] Tests indicated that at 12.5% outer fiber strain,[1] Inconel 600 U-bend specimens tested in high purity water at 650°F took a long incubation time (>12,000 hours) for the nucleation of an intergranular crack, longer time (13,000 hours) for >30% penetration and more than 18,000 hours to fail.

Although these test results are not directly applicable to the PWR steam generator tubing at Surry, they do confirm the observed operating experience that (1) a long incubation time is required to initiate intergranular cracking in Inconel 600 material, and (2) a high strain is required for crack propagation.

In this regard, the staff requested that the VEPCO address the following concern:

"Hourglassing" may continue and close the flow slots in the top support plate increasing the strain at the U-bend apex of the tubes in rows 2 and beyond.

In response to this concern, and to supplement plugging of row 1, VEPCO has installed stainless steel 304 alloy blocks in each of the six flow slots in the top support plate of all three Surry Unit No. 1 steam generators. These blocks prevent further closure of the flow slots and the inward displacement of the legs of the U-bends, thereby preventing further anticlastic straining at the U-bend apex of these tubes in rows 2 and beyond. As a result, intergranular stress corrosion cracking of those tubes at the U-bends in row 2 and beyond is not anticipated during near term (next year) normal operation. However, the flow slot blocking devices would cause: (1) an increase in strain in the support plate, (2) peripherial expansion of the support plate between wedge locations, (3) an increase of tube denting in the "hard spot" regions, and (4) additional bearing stresses on the wedges, wrapper, channel, and steam generator shell due to the peripherial expansion of the support plate. The net overall effect of flow slot blocking devices would be similar to complete closure of the flow slots. However, VEPCO had also increased selective tube plugging in the hard spot regions for the prevention of tube leaks at dented locations.

In this regard, the condition in the Indian Point Unit No. 2 steam generators has not approached these observed in either Surry Units,

as evidenced from the inspection results discussed later in this evaluation.

Support Plate Expansion

Continued growth of the magnetite in the tube-tube support plate annuli results in a non-uniform increase in strain in the support plates and corresponding in-plane expansion. In this regard, the staff requested that the VEPCO address the following concerns:

"1. Severe cracking of the support plate may result due to the continuing in-plane expansion of the support plate.
2. The rate of in-plane expansion in any support plate could increase the severity of tube denting in "hard spot" regions. Severe denting would restrain the tubes in the support plate and the plate may have a tendency to buckle or otherwise deform and thus exert additional bending loads on tubes.
3. With the closure of all the flow slots in any one support plate additional loads could be transmitted (due to the in-plane expansion of the plate) to the wedges, wrapper, channel spacer, tubes, and the steam generator vessel.
4. Thermal-hydraulic performance could be affected with the closure of all the flow slots in any support plate."

For the Indian Point Unit No. 2 steam generators, there is no evidence of any support plate cracking and thus these concerns are currently not applicable.

Anti-vibration Bar Fretting

On November 17, 1976 Southern California Edison Company (SCEC) reported to I&E, Region V, that, during the inspection of the San Onofre Unit No. 1 steam generators, excessive wear or mechanical fretting of anti-vibration bars was found in one of the steam generators. A failure of these bars could result in excessive flow induced vibration that might affect tube integrity, especially for those plants where the tube denting phenomenon was observed at the top support plate. Subsequent investigation revealed that the anti-vibration bar design of San Onofre Unit No. 1 and Connecticut Yankee is unique in comparison with other Westinghouse plants. Differences in the design are summarized as follows:

a. Materials - carbon steel for San Onofre Unit 1 and Connecticut Yankee; Inconel 600 for new models (44 and 51).

b. Bar Cross-section - 3/8 inch round bars; changed to square bars in the new models.

c. Clearances - (L-35 mils); was changed to (L-20 mils) for new models where L is the tube spacing.

d. Changes in V-bar configuration and spacing.

DISCUSSION

By letter dated May 6, 1977, the licensee submitted the results of the inspection performed on the Unit 2 steam generators, and requested NRC's concurrence for resumption for power operation.
During the next outage, which is planned for Spring, 1978, an inspection of the steam generators is presently scheduled.

Inspection Results

Eddy current examinations were performed on the hot leg sides of steam generators Nos. 21 and 22. Approximately 12% of the tubes in these two generators were inspected, which include some tubes in rows 2 through 5 and tubes in the peripheral hard spot regions. The extent and degree of tube denting was compared with that measured

during the first refueling outage in 1976. All tubes inspected near the inner tube lanes did not show any measurable change in the dent size. In the peripheral hard spot regions, eight (8) tubes in steam generator No. 21 and three (3) tubes in steam generator No. 22 did not allow passage of the 0.700 inch probe. Five of the eight tubes in generator No. 21 allowed passage of the 0.675 inch probe. The remaining six tubes (three in generator No. 21 and three in No. 22) allowed passage of the 0.610 inch probe.

Inspections of the flow slots in the first bottom tube support plate of steam generators Nos. 21 and 22 showed that the width of these flow slots remained unchanged from the previous inspection. The average flow slot "hourglassing" is about 1/8 inch, which is not significant.

The flow slots in the third and fourth support plates of steam generator Nos. 21 and 22 were examined for the first time during the April, 1977, outage. Some apparent "hourglassing" was observed photographically, but the amount of the "hourglassing" was judged to be similar to that measured in the bottom support plates.

In summary, the results of the inspections in the Indian Point Unit No. 2 steam generators did not indicate any strong evidence of the conditions observed in other severely affected PWR's currently under close NRC staff surveillance. The conditions of Unit No. 2 steam generators were essentially unchanged from the previous inspection, and no significant deterioration in these generators was found.

EVALUATION

We have reviewed the licensee's inspection reports, dated November 18, 1976, and May 6, 1977, and concur that there is no evidence of conditions that could lead to the U-bend failure due to excessive straining at the U-bend apex of rows 2 through 5 tubes in Indian Point Unit No. 2 steam generators. Although there is evidence of some denting of tubes in the peripheral hard spot regions, the amount of denting is not significant in comparison with those observed in other affected PWR's that have caused tube leaking.

With regard to the anti-vibration bar degradation problem revealed during the inspection of San Onofre Unit No. 1 steam generators, there is no reason to believe that similar problems will occur at Indian Point Unit No. 2 because there are basic differences in both the design and the material used. Anti-vibration bars in Indian Point Unit No. 2 steam generators are made of Inconel 600 instead of carbon steel, of square cross-section instead of round, and have smaller clearances than those originally employed at San Onofre Unit No. 1.

Since the time that Indian Point Unit No. 2 established AVT chemistry control, wastage and caustic stress corrosion cracking experience has been quite satisfactory. No substantial tube degradation from these corrosion mechanisms is expected to occur during normal operation.

Although the results of the inspection conducted by the licensee indicated only limited denting in those areas inspected, the staff believes that the inspection did not adequately sample areas which we have found in other facilities to be particularly susceptible to more severe denting. We cannot be certain from the licensee's inspection results alone that continued operation for the full period requested would be trouble free. However, the onset of further degradation would be accompanied by the occurrence of tube leakage. Consequently, we believe that there should be careful assessment of any leakage which might occur during operation. For this reason, the staff proposed and the licensee agreed to additional requirements for NRC approval of operation in the event any leak exceeds 1.0 gpm, or if leakage frequency should increase to two (2) or more tubes in a period of 20 days. With these restrictions, there is a high degree of assurance that degradation cannot proceed undetected to a degree which would entail safety concern.

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.

Dated: May 13, 1977

REFERENCES

1. H. A. Domian, et al. Effect of Microstructure on Stress Corrosion Cracking of Alloy 600 in High Purity Water. Corrosion, Vol. 33, P. 26, (January 1977).

2. R. L. Cowan and G. M. Gordon. Intergranular Stress Corrosion Cracking and Grain Boundary Composition of Fe-Ni-Cr Alloys, Preprint G-14 of paper presented at Stress Corrosion Cracking and Hydrogen Embrittlement of Iron Base Alloys Conference, Firminy, France (June 1973).

3. J. Blancnet, H. Coriou and et al. Influence of Various Parameters on Intergranular Cracking of Inconel 600 and X-750 in Pure Water at Elevated Temperature, Preprint G-13 of paper presented at Stress Corrosion Cracking and Hydrogen Embrittlement of Iron Base Alloys Conference, Firminy, France, (June 1973).

4. F. W. Pement and N. A. Granam. Stress Corrosion Cracking in High Purity Water, Scientific Paper 74-1B6-TUCOR-P1, Westinghouse Research Laboratories, (June 23, 1974).

5. Ph. Berge, H. D. Bui, J. R. Donati and D. Villard, Corrosion, Vol. 32, p. 357, (September 1976).

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-247

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 30 to Facility Operating License No. DPR-26, issued to the Consolidated Edison Company of New York, Inc. (the licensee), for operation of the Indian Point Nuclear Generating Unit No. 2 (the facility), located in Buchanan, Westchester County, New York. The amendment is effective as of its date of issuance.

The amendment requires an inspection of steam generators within 12 equivalent months of operation from May 13, 1977. The Technical Specifications for the facility have also been expanded to establish steam generator leakage limits.

The Commission has made appropriate findings as required by the Atomic Energy Act of 1954, as amended, 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.

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For further details with respect to this action, see (1) the licensee's submittal dated May 6, 1977, (2) Amendment No. 30 to License No. DPR-26 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 Hendrick Hudson Free Library, 31 Albany Post Road, Montrose, 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 13th day of May 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors