REGULATORY DOCKET FILE COPY

June 27, 1980

Distribution
Docket File 50-286
NRC PDR
Local PDR
TERA
NSIC
NRR Reading
ORB1 Reading
H. Denton
D. Eisenhut
T. Novak
S. Varga
L. Olshan
C. Parrish
I&E (5)
Attorney, OELD
B. Scharf (10)
B. Jones (4)
ACRS (16)
R. Diggs
C. Miles
C. Stephens

Docket No. 50-286

Mr. George T. Berry, President
and Chief Operating Officer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated January 28, 1980.

The amendment revises the license and the Technical Specifications to add a mid-cycle inspection of one steam generator and an end-of-cycle inspection of all four steam generators and incorporate new limiting conditions of operation for the steam generators.

You are requested to submit your detailed plan for the mid-cycle inspection at least 30 days prior to beginning the inspection.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

/s/

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 31 to DPR-64
2. Safety Evaluation
3. Notice of Issuance

~~8007180444~~

cc: w/enclosures
See next page

no legal objection

OFFICE	DL:ORB1	DL:ORB1	DL:AD:OR	OELD	DL:ORB1
SURNAME	LNOlshan:jb	CSParrish	TMNovak	J Moore	SAVarga
DATE	06/19/80	06/19/80	06/18/80	06/25/80	06/18/80

#10
44B19

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

June 27, 1980

Docket No. 50-286

Mr. George T. Berry, President
and Chief Operating Officer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated January 28, 1980.

The amendment revises the license and the Technical Specifications to add a mid-cycle inspection of one steam generator and an end-of-cycle inspection of all four steam generators and incorporate new limiting conditions of operation for the steam generators.

You are requested to submit your detailed plan for the mid-cycle inspection at least 30 days prior to beginning the inspection.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 31 to DPR-64
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

~~8007180~~ 046

cc: White Plains Public Library
100 Martine Avenue
White Plains, New York 10601

Mr. Charles M. Pratt
Assistant General Counsel
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Ms. Ellyn Weiss
Sheldon, Harmon and Weiss
1725 I Street, N.W., Suite 506
Washington, D. C. 20006

Dr. Lawrence R. Quarles
Apartment 51
Kendal at Longwood
Kennett Square, Pennsylvania 19348

Mr. George M. Wilverding
Licensing Supervisor
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Mr. P. W. Lyon, Senior Vice
President - Nuclear Generation
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Mr. J. P. Bayne, Resident Manager
Indian Point 3 Nuclear Power Plant
P. O. Box 215
Buchanan, New York 10511

Mr. J. W. Blake, Ph.D., Director
Environmental Programs
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Theodore A. Rebelowski
Resident Inspector
Indian Point Nuclear Generating
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

Joyce P. Davis, Esquire
Law Department
Consolidated Edison Company of
New York Inc.
4 Irving Place
New York, New York 10003

Jeffrey C. Cohen, Esquire
New York State Energy Office
Swan Street Building
CORE 1 - Second Floor
Empire State Plaza
Albany, New York 12223

Honorable George Begany
Mayor, Village of Buchanan
188 Westchester Avenue
Buchanan, New York 10511

cc: Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT STATION UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated January 28, 1980, 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.

~~8007180~~ 046

2. Accordingly, the Operating Licensing No. DPR-64 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, by changing the paragraph designated 2.J to 2.K, and by changing paragraphs 2.C(2) and 2.J to read as follows:

   2.C(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 31 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   2.J The plant shall be brought to the cold shutdown condition within nine equivalent months of operation from February 10, 1980, but in any event, no later than January 1, 1981. For the purpose of this requirement, equivalent operation is defined as operation with reactor coolant temperature greater than 350°F. An inspection of one steam generator shall be performed and Nuclear Regulatory Commission approval shall be obtained before bringing the reactor critical following this inspection. At the end of Cycle 3 operations, an inspection of all four steam generators shall be performed and Nuclear Regulatory Commission approval shall be obtained before bringing the reactor critical following this inspection.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 27,1980

ATTACHMENT TO LICENSE AMENDMENT NO. 31

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1-21	3.1-21
3.1-22	3.1-22
	3.1-22a
3.1-25	3.1-25
3.1-26	3.1-26
4.9-4	4.9-4
4.9-4a	4.9-4a
4.9-5	4.9-5
4.9-6	4.9-6
6-18	6-18

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 is determined that the indicated leak cannot be substantiated by direct observation or other indication.

2. If the leakage rate, excluding controlled leakage sources such as the Reactor Coolant Pump Controlled Leakage Seals and Leakage into Closed Systems, exceeds 1 gpm and the source of leakage is not identified within twenty-four hours of detection, the reactor shall be brought to hot shutdown within four hours. If the source of leakage is not identified within an additional twenty-four hours, the reactor shall be brought to a cold shutdown condition within the next twenty-four hours.

3. If the sources of leakage are identified and the results of the evaluation are that continued operation is safe, operation of the reactor with a total leakage, other than from controlled sources or into closed systems, not exceeding 10 gpm shall be permitted except as specified in 3.1.F.4 below.

4. If it is determined that leakage exists through a non-isolable fault which has developed in a Reactor Coolant System Component Body, pipe wall (excluding steam generator tubes), vessel wall or pipe weld, the reactor shall be brought to the cold shutdown condition within twenty-four hours.

5. If the total leakage, other than from controlled sources or into closed systems, exceeds 10 gpm, the reactor shall be placed in the hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours.

6. The reactor shall not be restarted following shutdown as per items 3.1.F.2, 3, 4, or 5, above, until the leak is repaired or until the problem is otherwise corrected.

7. 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, the Authority shall inform the NRC before the reactor is brought critical.

8. Primary to secondary leakage through the steam generator tubes shall be limited to 0.3 gpm (432 gpd) per steam generator and the total leakage through all four steam generators shall be limited to 1.0 gpm (1440 gpd). With any steam generator tube leakage greater than this limit the reactor shall be placed in the hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours.

9. 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 hot shutdown condition within four hours and the cold shutdown condition within an additional twenty-four hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation. If two steam generator tube leaks attributable to the tube denting phenomena are observed after the reactor is in cold shutdown Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

10. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different principles capable of detecting leakage into containment shall be in opertion, 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.

Measurement of the leakage rate to the containment atmosphere is also possible through humidity detection and condensation collection and measurement. However, it is expected that the containment activity method will give the initial indication of coolant leakage. The other methods will be employed primarily to confirm that leakage exists, to indicate the location of the leakage sources, and to measure the leakage rate.

As described above, the four reactor coolant leak detection systems are based on three 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 control.

Twenty-four hours is allowed from the time of leakage detection to identify the leakage source and to measure the leakage rate. This time period is required since identification and quantification of leakage sources of less than ten gallons per minute require a careful gathering and evaluation of data and/or a visual inspection of the reactor coolant system.

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 the limitation of steam generator leakage between the primary coolant system and the secondary coolant system. Cracks having a primary-to-secondary leakage less than 500 gallons per day during operation will have an adequate margin of safety against failure due to loads imposed by design basis accidents. The 500 gallon per day per steam generator limit is also consistent with the assumptions used to develop the Technical Specification limit on secondary coolant activity. Operating plants have demonstrated that primary-to-secondary leakage as low as 0.1 gpm will be detected. Leakage in excess of 432 gallons per steam generator or 1 gpm total for all four steam generators will require plant shutdown and an unscheduled eddy current inspection, during which the leaking tubes will be located and plugged.

References

FSAR Sections 11.2.3 and 14.2.4

4. Interval of Inspection

a. The first inservice inspection of steam generators should be performed after six effective full power months but not later than completion of the first refueling outage.

b. Subsequent inservice inspections should be not less than 12 or more than 24 calendar months after the previous inspection.

c. If the results of two consecutive inspections, not including the preservice inspection, all fall in the C-1 category, the frequency of inspection may be extended to 40-month intervals. Also, if it can be demonstrated through two consecutive inspections that previously observed degradation has not continued and no additional degradation has occurred, a 40-month inspection interval may be initiated.

B. Corrective Measures

All leaking tubes and defective tubes should be plugged.

C. Reports

1. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

2. The complete results of the steam generator tube inservice inspections shall be reported in writing on an annual basis for the period in which the inspection was completed per Specification 6.9.2.f. This report shall include:

   a. Number and extent of tubes inspected.

   b. Location and percent of wall-thickness penetration for each indication of an imperfection.

   c. Identification of tubes plugged.

3. Results of steam generator tube inspections which fall into Category C-3 of Table 4.9-1 require notification of the Commission within 15 days of this determination*. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

*Note - Table 4.9-1 requires NRC approval prior to startup in one case.

BASIS

Inservice inspection of steam generators is essential in order to monitor the integrity of the tubing and to maintain surveillance in the event that there is evidence of mechanical damage or progressive deterioration due to design, manufacturing errors, or chemical imbalance. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

An essentially 100% tube inspection was performed on each tube in every steam generator by eddy current techniques prior to service in order to establish a baseline condition for the tubing. This inspection was conducted under conditions and with equipment and techniques equivalent to those expected to be employed in the subsequent inservice inspections.

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 the limitation of steam generator leakage between the primary coolant system and the secondary coolant system. Cracks having a primary-to-secondary leakage less than 500 gallons per day during operation will have an adequate margin of safety against failure due to loads imposed by design basis accidents. Operating plants have demonstrated that primary-to-secondary leakage as low as 0.1 gpm will be detected. Leakage in excess of 432 gallons per day per steam generator or 1 gpm total through all four steam generators will require plant shutdown and an unscheduled eddy current inspection, during which the leaking and defective tubes will be located and plugged. The 500 gallon per day limit is also consistent with the assumptions used to develop the Technical Specification limit for secondary coolant activity.

Wastage-type defects are unlikely with the planned all volatile treatment (AVT) of secondary coolant. However, even if this type of defect occurs, the steam generator tube surveillance specification will identify steam generator tubes with impurifications having a depth greater than 40% of the 0.050 inch tube wall thickness as being unacceptable for continued service. The results of steam generator tube burst and collapse tests have demonstrated that tubes having wall thickness not less than 0.025 inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents.

A 10% allowance for tube degradation that may occur between inservice tube examinations added to the 40% tube plugging limit provides an adequate margin to assure that SG tubes acceptable for operation will not have a minimum tube wall thickness less than the acceptable 50% of normal tube wall thickness (i.e., 0.025 in) during the service lifetime of the tubes.

Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050 inch wall thickness.

d. Abnormal degradation of systems other than those specified in 6.9.1.7.c above designed to contain radioactive material resulting from the fission process. 7/

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Sealed source leakage on excess of limits (Specification 3.9)

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Primary coolant activity in excess of limits (Specification 3.1.D)

d. Seismic event analysis (Specification 4.10)

e. Inoperable fire protection and detection equipment (Specification 3.14)

f. The complete results of the steam generator tube inservice inspection (Specification 4.9.C)

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all source material of record.

i. Records of reactor tests and experiments.

7/ Sealed sources or calibration sources are not included under this item. Leakage of packing, gaskets, mechanical joints and seal welds within the limits for identified leakage set forth in technical specifications need not be reported under this item.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

INTRODUCTION

The results of the September 1979 steam generator inspection at Indian Point Unit No. 3 indicated an increase in the amount and extent of denting since the last inspection was performed in July 1978. These results were discussed during a meeting held on October 25, 1979 between representatives of the Power Authority of the State of New York (the licensee), Westinghouse, and the NRC, and were documented by the licensee in its submittal dated January 28, 1980. At the request of the NRC staff, the January 28, 1980 submittal includes a proposal for a license amendment to require a steam generator inspection midway through Cycle 3 operation, and proposed changes to the Indian Point 3 Technical Specifications to incorporate more stringent operating limits on primary to secondary leakage.

DISCUSSION

Previous Inspection

During the first refueling outage of Unit No. 3 in 1978, the first inservice inspection of the steam generators was performed on steam generators 33 and 34. This inspection revealed many dent indications with an average dent magnitude of 0.004 inches for tubes in the tubelane region. Away from the tubelane region, the dent indications were not of sufficient magnitude to be quantified. A substantial number of the dent indications occurred at the top tube support plates. No hourglassing (flow slot deformation) of the support plate flow slots was reported.

September 1979 Inspection Program

The original program for the steam generator inspection during the second refueling outage in September 1979 had called for multifrequency eddy current and gauging inspection of 491 tubes (9% sample), hot and cold legs and through U-bend, in steam generators 32 and 33. Tubes to be inspected included Row 1 and Row 2 tubes near the flow slots, Row 1 tubes between the flow slots, "hardspot" areas of the tube bundle periphery, and selected tubes throughout the center of the tube bundles. This program had been submitted for NRC staff review and had been modified to reflect ensuing staff comments.

Inspection of steam generator 32 indicated that the amount and extent of denting had progressed since the last inspection and flow slot hourglassing of the upper support plates was observed. Based upon these initial findings, it was decided to (1) plug all Row 1 tubes in each of the four steam generators, (2) complete the inspection of steam generators 32 and 33 as originally planned (with the exception that the eddy current inspection of steam generator 33 was performed using single frequency (400 KHZ) rather than multifrequency ECT), and (3) expand the program to include a gauging inspection of steam generators 31 and 34. The gauging inspection of steam generators 31 and 34 was performed on the hot leg side in the peripheral and patch plate areas and Row 2 tubes where denting could be expected to occur first.

The eddy current inspection of each steam generator involved the use of a 720 mil probe. If any tube did not pass this size probe, successively smaller probes were used until the size of the restriction was quantified. In all, five different size probes were used to determine the size of a restriction (720 mils, 700 mils, 650 mils, 600 mils, and 540 mils). In addition, all tubes adjacent to any tube which failed to pass a 650 mil probe were also subjected to eddy current testing with a 650 mil probe.

The implemented steam generator inspection included a photographic examination of the lower tube support plate and flow slots for all four steam generators using the hand holes above the tubesheet for access.

Sludge lancing was performed on all four steam generators to remove sludge from the areas above the tubesheet.

<u>INSPECTION RESULTS</u>

Eddy current inspection revealed no tubes with detectable indications (such as due to thinning or cracking). Significant tube restriction activity at support plate intersections (denting) was observed, however, as indicated in the following table:

HOT LEG

Steam Generator	No. of Tubes Inspected	No. of Tubes Restricting Probe Sizes of 720 mil	700 mil	650 mil	600 mil	540 mil
31	488	254	73	5	2	1
32	537	288	161	15	3	0
33	682	294	130	19	9	1
34	498	243	127	18	3	1
		COLD LEG				
32	501	74	5	0	0	0
33	526	107	41	8	1	0

Photographs of the flow slots for the lower two support plates revealed that "hourglassing" had occurred and that some support plate cracking had occurred at the flow slots. Flow slot closures ranged to a maximum of 0.55 inch with an average value of 0.37 inch. The average rate of hourglassing was calculated to be 0.033 inch per month since the previous inspection.

Tube plugging

All 92 Row 1 tubes in each of the four steam generators were preventively plugged as a result of the observed flow slot hourglassing in the lower support plates and the potential for hourglassing in the upper support plate. This was done to preclude leaks and potential tube rupture due to cracks at the apex of the tube U-bends which could be induced by the support displacement of the legs of the U-bends because of the hourglassing of the flow slots. This phenomenon was responsible for an 80 GPM tube rupture event at Surry Unit 1 in 1976 affecting a Row 1 tube at the apex of the U-bend. Laboratory examinations and analyses of tubes from units which have experienced hourglassing, and operating experience indicate the Row 1 tubes to be the most susceptible to this U-bend cracking phenomenon. U-bend cracks or leaks due to this phenomenon have not been observed at any unit to date beyond Row 1 tubes.

In addition to Row 1 tubes, all tubes restricting passage of a 650 mil probe were plugged, including five (5) tubes in steam generator 31, 15 tubes in steam generator 32, 21 tubes in steam generator 33, and 18 tubes in steam generator 34. These numbers do not include 10 tubes which were inadvertantly plugged. The total plugging count to date at Indian Point Unit No. 3 is 104 tubes (3.2%) for steam generator 31, 115 tubes (3.5%) for steam generator 32, 117 tubes (3.6%) for steam generator 33, and 119 tubes (3.7%) for steam generator 34.

PROPOSED LICENSING AMENDMENT AND TECH SPEC CHANGES

In view of concerns expressed by the NRC staff during the October 25, 1979 meeting regarding the apparent high rate of denting at Indian Point Unit No. 3, the licensee has proposed a change to its operating license to require a mid-cycle inspection of one steam generator during the 18 month (approximately) period of Cycle 3 operation, and an inspection of all four steam generators at the conclusion of the cycle. NRC approval would be required before critical operation could be resumed following both the mid-cycle and end of cycle inspections.

As requested by the NRC staff, the licensee has proposed changes to the Technical Specification limits on primary to secondary leakage through the steam generators. With the proposed changes, the leak rate limit for any one steam generator would be reduced from 0.348 gpm to 0.30 gpm, beyond which the unit would be required to shutdown for steam generator inspection and repair. The proposed changes include an added requirement to shutdown for steam generator inspection and repair if any two separate tubes are found to leak during any 20 day period, regardless of the leakage level of each tube. Whenever the reactor is shutdown, or a steam generator removed from service, to investigate a steam generator leak and/or to repair a leaking tube, the Technical Specifications would require that the NRC be notified before the reactor is brought critical.

The remaining changes to the Technical Specifications proposed by the licensee involve clarifications in the reporting requirements for steam generator inspections. We have reviewed these changes and consider them not to change, in a substantive sense, current reporting requirements.

EVALUATION

Based upon previous denting related tube leak occurrences on December 7, 1978 and March 20, 1979, and the finding of widespread tube restrictions and hourglassing of the tube support plate flow slots during the most recent inspection, the staff considers Indian Point Unit No. 3 to be the latest among 10 operating PWR units which have experienced moderate to extensive denting of the steam generator tubes. The number of tubes restricting passage of a .650 inch probe or less is still small compared to the number of affected tubes in more severely degraded units such as Indian Point Unit No. 2, Turkey Point Unit Nos. 3 and 4, and Surry Unit Nos. 1 and 2 (prereplacement steam generators). However, the average rate of flow slot hourglassing, calculated to be 0.033 inches per month since the previous inspection, is high compared to what has been observed at other units and suggests that the denting phenomenon may be developing at a significant rate at Indian Point Unit No. 3.

As requested by the staff, the licensee has proposed a change to its operating license to require a mid-cycle inspection of one steam generator during Cycle 3 operation, and an inspection of all four steam generators at the end of the cycle. Under this proposal, NRC approval would be required before critical operation could be resumed following both the mid-cycle and end of cycle inspections. We find that the licensee's proposal provides adequate provision for monitoring the rate of denting and for establishing, on a timely basis, the need for additional licensing actions (e.g., more restrictive preventive plugging criteria). The amendment requires that Unit No. 3 be required to shutdown for the mid-Cycle 3 inspection within nine (9) equivalent full power months from the start of Cycle 3 operation, but not after January 1, 1981. For purposes of this SER, equivalent full power operation is defined as operation with primary coolant temperature greater than 350°F.

It should be noted that NRC approval to resume equivalent full power operation following the mid-cycle and end of Cycle 3 steam generator inspections will be contingent upon the adequacy of the inspections performed in view of the results obtained, and also the adequacy of the implemented preventive plugging program to support continued operation to the next scheduled steam generator inspection. In addition, approval to resume full power operation following the mid-cycle inspection of one steam generator will be contingent on whether the inspection results adequately justify not performing a mid-cycle inspection of the other three steam generators.

The more restrictive primary to secondary leakage rate limits proposed by the licensee are consistent with those currently in effect at other more severely degraded units. The proposed changes provide additional assurance that the occurrence of a through wall crack during operation will be detected and appropriate corrective action will be taken such that an individual crack will not become unstable and burst under normal operating, transient, or accident conditions.

In conclusion, we find that the proposed changes to the Indian Point Unit No. 3 operating license and Technical Specifications, as identified in the licensee's submittal dated January 28, 1980, will provide reasonable assurance of continued safe operation of the unit.

Environmental Consideration

We have determined that the amendments do 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 amendments involve 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 these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 27,1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-286

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 31 to Facility Operating License No. DPR-64, issued to the Power Authority of the State of New York (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 3 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment requires a mid-cycle inspection of one steam generator, an end-of-cycle inspection of all four steam generators, and incorporates new limiting conditions of operation for the steam generators.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act 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.

For further details with respect to this action, see (1) the application for amendment dated January 28, 1980, (2) Amendment No. 31 to License No. DPR-64, 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 White Plains Public Library, 100 Martine Avenue, White Plains, 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 Licensing.

Dated at Bethesda, Maryland, this 27th day of June, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing