BEFORE THE UNITED STATES

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
POWER AUTHORITY OF THE STATE OF NEW YORK	)	Docket No. 50-286
Indian Point 3 Nuclear Power Plant	)	

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the Nuclear Regulatory Commission, the Power Authority of the State of New York, as holder of Facility Operating License No. DPR-64, hereby applies for an Amendment to the Technical Specifications contained in Appendix A of this license.

This application proposes a one time deferral of the Type C test (local leak rate test) for Residual Heat Removal (RHR) containment isolation valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870. The leak rate test for these valves would be deferred until the next refueling outage, currently scheduled for the spring of 1996.

The proposed changes to the Technical Specifications are included as Attachment II to this application. The Safety Evaluation is included as Attachment III.

POWER AUTHORITY OF THE
STATE OF NEW YORK

By
William J. Cahill, Jr.
Executive Vice President and
Chief Nuclear Officer
Nuclear Generation

STATE OF NEW YORK
COUNTY OF WESTCHESTER

Subscribed and Sworn to before me
this 29th day of September, 1994

Geraldine Strand
Notary Public

GERALDINE STRAND
Notary Public, State of New York
No. 4991272
Qualified in Westchester County
Commission Expires Jan. 27, 1996

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ATTACHMENT I TO IPN-94-122

JUSTIFICATION FOR SCHEDULAR EXEMPTION FROM PLANT SPECIFIC REQUIREMENTS ASSOCIATED WITH 10 CFR 50, APPENDIX J, SECTION III.D.3 FOR RESIDUAL HEAT REMOVAL CONTAINMENT ISOLATION VALVES AC-732, AC-741, AC-MOV-743, AC-MOV-744, AND AC-MOV-1870

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

JUSTIFICATION FOR SCHEDULAR EXEMPTION FROM PLANT SPECIFIC REQUIREMENTS ASSOCIATED WITH 10 CFR 50, APPENDIX J, SECTION III.D.3 FOR RESIDUAL HEAT REMOVAL CONTAINMENT ISOLATION VALVES AC-732, AC-741, AC-MOV-743, AC-MOV-744, AND AC-MOV-1870

In accordance with 10 CFR 50.12(a), The New York Power Authority requests a one-time schedular exemption from an Indian Point 3 Nuclear Power Plant (IP3) specific requirement associated with 10 CFR 50, Appendix J, Section III.D.3. The exemption requested would allow for the deferral of the Type C test (local leak rate test) of Residual Heat Removal containment isolation valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870 until the next scheduled refueling outage. This requires a temporary waiver of the 30 month maximum surveillance interval for the Type C tests of these valves.

10 CFR 50, Appendix J, Section III.D.3 states:

> Type C tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than 2 years.

The Authority received an exemption from the above requirement for IP3. The exemption was granted by an NRC letter to R.E. Beedle, dated February 19, 1993. The exemption states:

> The Power Authority of the State of New York is exempt from the requirement of 10 CFR Part 50, Appendix J, Paragraph III.D.3, in that the interval for Type C tests may be extended greater than 2 years but in no case greater than 30 months for the Indian Point Nuclear Generating Unit No. 3.

The Residual Heat Removal (RHR) containment isolation valves were previously tested during the last refueling outage (the 8/9 refueling outage). The Type C tests were performed on May 7, 1992 for valve AC-732, on June 5, 1992 for valve AC-741, and on May 25, 1992 for valves AC-MOV-743, AC-MOV-744, and AC-MOV-1870. The Authority began cycle 9 operation in August 1992. Approximately six months after startup from the 8/9 refueling outage, the Authority brought IP3 to cold shutdown and began an unplanned outage (the current Restart and Continuous Improvement Outage). The Restart and Continuous Improvement (RCIP) Outage has been an extended non-refueling outage. Startup from the RCIP Outage is currently expected to begin in early 1995. After startup from the current outage, the plant will run until the next refueling outage (the 9/10 refueling outage) which is currently scheduled to begin in the spring of 1996. As a result of the extended RCIP Outage, the 30 month leak rate test interval for valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870 will be reached in November 1994 and December 1994 (approximately 16 months prior to the scheduled 9/10 refueling outage). This exemption would permit a deferral in the performance of the Type C test of the RHR containment isolation valves until the 9/10 refueling outage.

The leak rate tests performed to satisfy 10 CFR Part 50-Appendix J-Type C testing for the subject containment isolation valves also satisfy the leak rate testing requirements identified in the Inservice Testing (IST) Program (transmitted to the NRC by IPN-93-139, dated November 9, 1993). The IST Program states that the frequency for leak rate testing of these valves is every two years. For the same reasons discussed below, the IST interval for leak rate testing of the subject valves would require an extension to allow the valves to be tested during the next refueling outage.

JUSTIFICATION FOR EXEMPTION

10 CFR 50.12(a) states that the Nuclear Regulatory Commission (NRC) may grant exemptions from the requirements of the regulations contained in 10 CFR 50 provided that:

- The exemption is authorized by law;
- The exemption does not present an undue risk to the public health and safety;
- The exemption will not endanger the common defense and security;
- Special circumstances are present as defined in 10 CFR 50.12(a)(2).

The Nuclear Regulatory Commission has already granted an exemption from the requirements of 10 CFR 50, Appendix J, Section III.D.3 for IP3 and has issued plant specific requirements associated with the regulation. This application requests a temporary schedular exemption from those plant specific requirements for certain containment isolation valves. Therefore, the same criteria for granting exemptions to NRC regulations will be addressed in this application for revision to the plant specific requirements associated with 10 CFR 50, Appendix J, Section III.D.3. Section IV, below, describes the special circumstances that necessitate this request for exemption.

I. The requested exemption is authorized by law.

The Nuclear Regulatory Commission is authorized by law to grant this exemption. The NRC has granted similar exemptions to other nuclear power plants including the James A. FitzPatrick nuclear power plant which received a one-time exemption from the requirement to perform Type C testing within the 2 year testing interval for the shutdown cooling isolation valves.

II. The requested exemption does not present an undue risk to the public health and safety.

As part of the effort to accommodate operation on a 24-month fuel cycle, the results of all containment isolation valve leakage tests that were performed through 1990 were evaluated to determine whether the containment leakage rate would be maintained within acceptable limits with an interval increase of up to 30 months. Based on this evaluation, the Authority

concluded that the leakage failures, except for one valve which was replaced in 1990, were random and nonrecurring. Therefore, the Authority concluded that these failures were not indicative of a poor performance trend and that there was no evidence that the containment leakage was a function of time. This conclusion was stated in the NRC's safety evaluation associated with License Amendment 129, issued on April 9, 1993. Therefore, there is reasonable assurance that the containment leakage rate would be maintained within acceptable limits with the deferral of the leakage rate tests for the RHR containment isolation valves until the 9/10 refueling outage.

Below is a specific evaluation of the safety implications of the proposed technical specification change for each of the three RHR lines associated with the change.

RHR Loop Outlet Line - Valves AC-741 and AC-MOV-744

On the RHR loop outlet line, valve AC-741 is a check valve inside containment and valve AC-MOV-744 is a double disc gate valve outside containment. Valve AC-741 is designed to be closed whenever the RHR system is not in service. The only time that valve AC-MOV-744 is required to be closed for containment isolation is in certain post-accident recirculation modes. After AC-MOV-744 is closed, nitrogen injection flow from the Isolation Valve Seal Water System (IVSWS) is manually initiated by procedure. The IVSWS ensures that leakage past the containment isolation valve is minimized by providing a seal at the valve. When a double disc gate valve, such as AC-MOV-744, is closed, the upstream and downstream discs are forced against their respective seats, therefore, redundant isolation barriers are provided in a single valve. In the case of valve AC-MOV-744, the nitrogen from the IVSWS is injected through the valve bonnet and pressurizes the space between the two valve discs. The nitrogen pressure in excess of the potential accident pressure minimizes outleakage past the first isolation point of the valve.

Therefore, there is reasonable assurance that reactor coolant water will not leak past the outer containment isolation valve (valve AC-MOV-744) because there are redundant isolation barriers (valve AC-741 and the two discs of valve AC-MOV-744) and because any further hypothesized leakage past these isolation barriers would be minimized by the seal provided by the IVSWS. Additionally, on a monthly basis, valve AC-741 is exercised partially open when the Reactor Coolant System is hot and the steam generators are being used as the primary means of heat removal, and is exercised fully open during plant shutdowns when the RHR system is being used to remove reactor decay heat. Although these tests do not verify valve closure or leak tightness, these monthly surveillances would indicate certain valve mechanical problems.

Further, a periodic surveillance that measures the leakage from RHR and Safety Injection System components that are located outside containment and are used during the recirculation phase of a design basis accident is performed. This surveillance is performed in accordance with Technical Specification 4.4.I. Depending on the results of the surveillance, corrective actions are taken to ensure that the leakage from these components does not exceed two gallons per hour. Recirculation system leakage is limited to two gallons per hour or less to ensure that off-site exposures due to this leakage are limited to

insignificant levels. Therefore, even if there was significant leakage past the RHR containment isolation valves, the RHR system components located outside containment would not leak significant amounts of the radioactive reactor coolant water to the Primary Auxiliary Building. Therefore, significant leakage past the RHR containment isolation valves would not significantly affect off-site exposures.

The most recent leakage rate tests were performed on May 25, 1992, for valve AC-MOV-744, and on June 5, 1992, for valve AC-741. The leak rates of the valves were well within the acceptance criteria of the tests.

RHR Miniflow Line - Valves AC-MOV-743 and AC-MOV-1870

On the RHR miniflow line, valve AC-MOV-743 is a gate valve and AC-MOV-1870 is a globe valve. Both isolation valves are located outside containment. The only time that valves AC-MOV-743 and AC-MOV-1870 are required to be closed for containment isolation is in certain post-accident recirculation modes. After the valves are closed, nitrogen gas from the high pressure nitrogen supply of the IVSWS is injected into the pipe between the two valves. The nitrogen pressure in excess of the potential accident pressure minimizes outleakage past the first isolation valve (AC-MOV-1870).

Therefore, there is reasonable assurance that reactor coolant water will not leak past the outer containment isolation valve (valve AC-MOV-743) because there are redundant isolation barriers (valve AC-MOV-1870 and valve AC-MOV-743) and because any further hypothesized leakage past these isolation barriers would be minimized by the seal provided by the IVSWS. Additionally, valves AC-MOV-743 and AC-MOV-1870 are full stroke tested during cold shutdowns. This surveillance ensures that the valves are mechanically operable. (This surveillance was last performed on September 19, 1994).

Further, a periodic surveillance that measures the leakage from RHR and Safety Injection System components that are located outside containment and are used during the recirculation phase of a design basis accident is performed. This surveillance is performed in accordance with Technical Specification 4.4.I. Depending on the results of the surveillance, corrective actions are taken to ensure that the leakage from these components does not exceed two gallons per hour. Recirculation system leakage is limited to two gallons per hour or less to ensure that off-site exposures due to this leakage are limited to insignificant levels. Therefore, even if there was significant leakage past the RHR containment isolation valves, the RHR system components located outside containment would not leak significant amounts of the radioactive reactor coolant water to the Primary Auxiliary Building. Therefore, significant leakage past the RHR containment isolation valves would not significantly affect off-site exposures.

The most recent leakage rate test for these valves was performed on May 25, 1992. The leak rates of the valves were well within the acceptance criteria of the test.

RHR Loop Shutdown Inlet Line - Valve AC-732

On the RHR loop shutdown inlet line, valve AC-732 is a double disc gate valve outside containment. This valve is required to be in the closed position during normal plant operation and after a Loss of Coolant Accident (LOCA). The valve is only opened when the RHR system is needed to remove reactor decay heat. Like valve AC-MOV-744, nitrogen gas from the high pressure nitrogen supply of the IVSWS can be injected through the bonnet of valve AC-732. The nitrogen pressurizes the space between the two valve discs. The nitrogen pressure in excess of the potential accident pressure minimizes outleakage past the first isolation point of the valve.

Additionally, upstream of valve AC-732 are stop valves AC-MOV-730 and AC-MOV-731. Valves AC-MOV-730 and AC-MOV-731 are in the closed position during normal plant operation and during post-LOCA time periods. The valves are only opened when the RHR system is needed to remove reactor decay heat. The valves are pressure interlocked to prevent opening when Reactor Coolant System pressure is above the RHR system design pressure. This operating configuration substantially minimizes the possibility of gross leakage through valve AC-732.

Therefore, there is reasonable assurance that reactor coolant water will not leak past containment isolation valve AC-732 because there are redundant means of isolating the reactor coolant system (valve AC-MOV-730 and valve AC-MOV-731), redundant barriers for containment isolation (the two discs of valve AC-732), and any further hypothesized leakage past the containment isolation barriers would be minimized by the seal provided by the IVSWS.

Further, a periodic surveillance that measures the leakage from RHR and Safety Injection System components that are located outside containment and are used during the recirculation phase of a design basis accident is performed. This surveillance is performed in accordance with Technical Specification 4.4.I. Depending on the results of the surveillance, corrective actions are taken to ensure that the leakage from these components does not exceed two gallons per hour. Recirculation system leakage is limited to two gallons per hour or less to ensure that off-site exposures due to this leakage are limited to insignificant levels. Therefore, even if there was significant leakage past the RHR containment isolation valves, the RHR system components located outside containment would not leak significant amounts of the radioactive reactor coolant water to the Primary Auxiliary Building. Therefore, significant leakage past the RHR containment isolation valves would not significantly affect off-site exposures.

The most recent leakage rate test for valve AC-732 was performed on May 7, 1992. The leak rate of the valve was well within the acceptance criteria of the test.

III. The requested exemption is consistent with the common defense and security.

The common defense and security are not affected by this request for exemption.

IV. Special circumstances are present which necessitate the request for an exemption.

Three of the special circumstances presented in 10 CFR 50.12(a)(2) apply to this exemption:

1. Circumstance (ii) states: "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

   The underlying purpose of Section III.D.3 of Appendix J to 10 CFR Part 50 is to provide an interval short enough to prevent serious deterioration from occurring between tests and long enough to permit testing to be performed during regular plant outages. In recognition of the fact that many containment isolation valves can not be tested at power and that certain valves can not be tested during non-refueling outages, the NRC has granted schedular exemptions to Section III.D.3 for many licensees including the James A. FitzPatrick Nuclear Power Plant.

   For containment isolation valves that can not be tested at power or for containment isolation valves where testing involves unreasonable risk to personnel and equipment, the increased confidence in containment integrity following successful testing is not significant enough to justify the risk to plant safety or the hardships associated with performing the test within the test interval.

   Any potential incremental benefit of performing the leak rate tests for valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870 within the 30-month requirement would not be sufficient to offset the increased occupational radiation exposure associated with testing, the risk to plant safety associated with removing the primary method of decay heat removal from service (as described below), and the undue financial burden of placing the plant in the refueling configuration.

2. Circumstance (iii) states: "Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, ..."

   During non-refueling outages, such as the current Restart and Continuous Improvement Outage, fuel typically remains in the reactor with the reactor head in place. In this condition, the RHR system is the principal system used to remove reactor decay heat. Since, typically, only one RHR pump and one RHR heat exchanger is required to remove reactor decay heat, the RHR system, with two RHR pumps and two RHR heat exchangers, has sufficient redundancy to allow one pump or one heat exchanger to be out of service. The steam generators are a backup means for decay heat removal.

   However, when testing RHR containment isolation valves AC-732, AC-741, AC-MOV-743, AC-MOV-744 or AC-MOV-1870, current procedures require that the entire RHR system be removed from service. This would leave the steam generators as the only available means for decay heat removal. Therefore, there would be no redundant means available for decay heat removal. Additionally, the steam generators do not provide the forced circulation that is required to prevent boron dilution. Therefore, with fuel in the reactor

and the reactor head in place, it is not sufficient to have the steam generators as the only available means for decay heat removal.

To provide sufficient core protection during testing of the RHR containment isolation valves current testing procedures would require that the plant be in the refueling configuration; i.e., reactor head removed and the reactor cavity flooded up. This permits the removal of the RHR system from service and testing of the RHR containment isolation valves.

An exemption from the 30-month containment isolation valve leak rate test interval for valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870 will preclude the need to place the plant in the refueling configuration prior to the next scheduled refueling outage. Using Authority resources to put the plant in the refueling configuration is not practical solely for the purpose of testing the RHR containment isolation valves. Additionally, placing the plant in the refueling configuration will significantly increase occupational radiation exposures. For these reasons, compliance with the regulation would result in undue hardship and costs that are significantly in excess of those contemplated when the regulation was adopted.

3. Circumstance (v) states: "The exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation."

   The requested exemption is temporary since it provides relief from the 30-month maximum surveillance interval only until the cycle 9/10 refueling outage. Type C testing of the subject valves would be performed during the 9/10 refueling outage, currently scheduled to begin approximately 16 months after the 30-month interval limit will be reached. The Authority has made a good faith effort to comply with the regulations considering the fact that the only valves for which relief is being requested are the valves whose testing procedure would require removing the RHR system from service.

CONCLUSION

The Authority concludes that this one-time schedular exemption is warranted under the provisions of 10 CFR 50.12, in that it does not present an undue risk to the public health and safety, and several "special circumstances" are present. There is a high degree of confidence that the affected components will not significantly degrade during the extended operating interval. Moreover, any potential incremental benefit of performing the tests within the 30-month requirement would not be sufficient to offset the increased occupational radiation exposure associated with testing, the risk to plant safety associated with the removal from service of the primary method of decay heat removal, and the undue financial burden of placing the plant in the refueling configuration.