ATTACHMENT II TO IPN-94-122

TECHNICAL SPECIFICATION CHANGES

ASSOCIATED WITH

LEAK RATE TESTING INTERVAL 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

E. Containment Isolation Valves

1. Tests and Frequency

a. Isolation valves in Table 4.4-1 shall be tested for operability at intervals no greater than 30 months (24 months + 25%).

b. Isolation valves in Table 4.4-1 which are pressurized by the Weld Channel and Penetration Pressurization System shall be leakage tested as part of the Weld Channel and Penetration Pressurization System Test at intervals no greater than 30 months (24 months + 25%).

c. Isolation valves in Table 4.4-1 which are pressurized by the Isolation Valve Seal Water System shall be tested at intervals no greater than 30 months (24 months + 25%)* as part of an overall Isolation Valve Seal Water System Test.

d. Isolation valves in Table 4.4-1 which are not pressurized will be tested at intervals no greater than 30 months (24 months + 25%)*.

e. Isolation valves in Table 4.4-1 shall be tested with the medium and at the pressure specified therein.

2. Acceptance Criteria

a. The combined leakage rate for the following shall be less than 0.5 L_a: isolation valves listed in Table 4.4-1 subject to gas or nitrogen pressurization testing, air lock testing as specified in D.1, portions of the sensitive leakage rate test described in C.1 which pertain to containment penetrations and double-gasketed seals.

b. The leakage rate into containment for the isolation valves sealed with the service water system is 0.36 gpm per fan cooler.

c. The leakage rate for the Isolation Valve Seal Water System shall not exceed 14,700 cc/hr.

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* For fuel cycle 9, leakage testing of Containment Isolation Valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870 may be deferred until the 9/10 refueling outage.

ATTACHMENT III TO IPN-94-122

SAFETY EVALUATION OF TECHNICAL SPECIFICATION CHANGES

ASSOCIATED WITH

LEAK RATE TESTING INTERVAL FOR RESIDUAL HEAT REMOVAL

CONTAINMENT ISOLATION VALVES

AC-732, AC-741, AC-MOV-743, AC-MOV-744, AND AC-MOV-1870

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SAFETY EVALUATION OF TECHNICAL SPECIFICATION CHANGES ASSOCIATED WITH LEAK RATE TESTING INTERVAL FOR RESIDUAL HEAT REMOVAL CONTAINMENT ISOLATION VALVES AC-732, AC-741, AC-MOV-743, AC-MOV-744, AND AC-MOV-1870

Section I - Description of Changes

This application for amendment to the Indian Point Unit 3 (IP3) Technical Specifications proposes to revise Section 4.4 of Appendix A of the Operating License. The proposed revision to Technical Specification 4.4.E.1 would allow a one time extension to the 30 month interval requirement for leak rate testing of Residual Heat Removal (RHR) containment isolation valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870.

Section II - Evaluation of Changes

In accordance with Technical Specification 4.4.E.1, containment isolation valves are currently required to be leak rate tested at intervals no greater than 30 months. As a result of the current extended Restart and Continuous Improvement Outage, this 30 month interval will be exceeded for some valves before the next refueling outage, scheduled for the spring of 1996. Except for RHR containment isolation valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870, all containment isolation valves can be tested prior to exceeding the 30 month requirement. However, in order to perform the leak rate tests for the RHR containment isolation valves, current procedures require that the RHR system be taken out of service. The current Restart and Continuous Improvement Outage is a non-refueling outage, therefore, it would be impractical to remove the RHR system from service because the system is needed to remove reactor decay heat. Therefore, the Authority proposes to defer the leak rate tests for valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870 until the next refueling outage, when the RHR system can be safely removed from service. Since the next refueling outage is scheduled to begin after the 30 month testing interval would have already been exceeded, proposed changes to Technical Specification 4.4.E.1 would allow a one time extension to the 30 month interval for leak rate testing of RHR valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870.

The leak rate tests performed to satisfy Technical Specification 4.4.E.1 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.

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, under current procedures, the plant needs to 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.

The proposed change to Technical Specification 4.4.E.1 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.

The RHR containment isolation valves for which an extension to the 30 month interval for leak rate testing is being proposed are located on three lines which penetrate containment. Valves AC-741 and AC-MOV-744 are on the RHR loop outlet line. Valves AC-MOV-743 and AC-MOV-1870 are on the RHR miniflow line. Valve AC-732 is on the RHR loop shutdown inlet line (line from hot leg 32 of the Reactor Coolant System).

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 means of isolation (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.

<u>Section III - No Significant Hazards Evaluation</u>

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response:

The proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change is limited to a one-time extension of the containment isolation valve leak rate test for RHR valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870. The change does not introduce any new modes of plant operation, make any physical changes, or alter any operational setpoints. Therefore, the change does not degrade the performance of any safety system assumed to function in the accident analysis. There is reasonable assurance that the extension will not result in a significant increase in valve leakage considering that a review of containment isolation valve leakage rate test results through 1990 showed that all leakage failures, except for one valve which was replaced in 1990, were random and nonrecurring. Additionally, each of the three RHR lines associated with valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870 has redundant isolation barriers and is supplied by the IVSWS which would minimize any leakage past the isolation barriers. Further, due to the periodic surveillance that ensures that leakage from RHR components located outside containment does not exceed two gallons per hour, even if significant leakage past the RHR containment isolation valves occurred, this would not significantly affect off-site exposures.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change does not introduce new accident initiators or failure mechanisms since the change does not alter the physical characteristics of any plant system or component. The change is limited to a one-time extension to the leak rate test interval for RHR valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed amendment does not involve a significant reduction in a margin of safety. There is reasonable assurance that the extension will not result in a significant increase in valve leakage since a review of containment isolation valve leakage rate test results through 1990 showed that all leakage failures, except for one valve which was replaced in 1990, were random and nonrecurring. Additionally, each of the three RHR lines associated with valves AC-732, AC-741, AC-MOV-743, AC-MOV-744, and AC-MOV-1870 has redundant isolation barriers and is supplied by the IVSWS which would minimize any leakage past the isolation barriers. Further, due to the periodic surveillance that ensures that leakage from RHR components located outside containment does not exceed two gallons per hour, even if significant leakage past the RHR containment isolation valves occurred, this would not significantly affect off-site exposures.

<u>Section IV - Impact of Changes</u>

These changes will not adversely affect the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or SER Conclusions
Overall Plant Operations and the Environment

<u>Section V - Conclusions</u>

The incorporation of these changes: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety

Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any technical specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VII - References

a) IP3 FSAR
b) IP3 SER