ATTACHMENT II to JPN-94-001

PROPOSED TECHNICAL SPECIFICATION CHANGE

SHUTDOWN COOLING ISOLATION VALVE LEAK RATE TESTING

JPTS-93-022

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

4.7 (cont'd)

(5) Type C test.

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

(6) Other leak rate tests specified in Section 4.7.d shall be performed during each reactor shutdown for refueling but in no case at intervals greater than two years.

f. Containment modification*

Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. The measured leakage from this test shall be included in the test report. The acceptance criteria as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

* In accordance with an exemption from 10 CFR 50 Appendix J, a Type A, B, or C test is not required for the replacement of piping and welds which constitute the Core Spray System minimum flow lines (3"-W23-152-7A, B) during the 1993 maintenance outage.

** In accordance with an exemption from 10 CFR 50 Appendix J, the Type C test of the shutdown cooling isolation valves (10MOV-17 and 10MOV-18) may be deferred until refueling outage Reload 11/ Cycle 12.

ATTACHMENT III to JPN-94-001

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGE

SHUTDOWN COOLING ISOLATION VALVE LEAK RATE TESTING

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Attachment III to JPN-94-001

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGE SHUTDOWN COOLING ISOLATION VALVE LEAK RATE TESTING

I. DESCRIPTION OF THE PROPOSED CHANGE

The proposed change to the James A. Fitzpatrick Technical Specifications revises Specification 4.7.A.2.e(5) on page 174 as follows:

A double asterisk (**) is inserted after Specification 4.7.A.2.e(5) and the following note is added to the bottom of the page:

> ** In accordance with an exemption from 10 CFR 50, Appendix J, the Type C test of the shutdown cooling isolation valves (10MOV-17 and 10MOV-18) may be deferred until refueling outage Reload 11/ Cycle 12.

II. PURPOSE OF THE PROPOSED CHANGE

Technical Specification 4.7.A.2.e(5) states:

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

The proposed change to the James A. FitzPatrick Technical Specification would permit a deferral in the performance of the Type C test (local leak rate test) of the shutdown cooling isolation valves beyond the two year maximum surveillance interval to the next refueling outage. The change conforms with the accompanying exemption request to 10 CFR 50, Appendix J, Section III.D.3.

The shutdown cooling isolation valves were previously tested during the last refueling outage (Reload 10/Cycle 11). This was an extended outage that began in November 1991 and ended in January 1993. The Type C tests on the subject valves were performed on May 30, 1992 for the outboard shutdown cooling isolation valve (10MOV-17), and June 5, 1992 for the inboard shutdown cooling isolation valve (10MOV-18). Subsequent delays in the outage resulted in these tests being performed significantly in advance of the start of operating cycle 11 (more than seven months prior to the end of the outage). As a result, the two year test interval will be reached for these valves (May 30 1994 / June 5, 1994) six to seven months prior to the next refueling outage in November 1994.

The shutdown cooling isolation valves are used to isolate the single suction line from the reactor coolant system to the Residual Heat Removal (RHR) system pumps. The only effective means of removing reactor core decay heat is with the shutdown cooling mode of the RHR system. This requires both isolation valves to be in the open position. The shutdown cooling mode of the RHR system must be removed from service for 24 hours to perform a local leak rate test (Type C). This is the time required to tag-out the system, drain the line, perform the test, refill the line, and return the system to service. To avoid overheating the reactor coolant system with the shutdown cooling mode inoperable, one of the following two conditions must exist:

1. The reactor needs to be shutdown for several months to permit sufficient reduction in decay heat levels for use of an alternate shutdown cooling method without placing the plant in the refueling configuration. The alternate cooling method with the highest heat removal capacity, as identified in a plant procedure (Shutdown Procedure ODSO-32), is the Reactor Water Cleanup system in the blowdown mode. However, the reactor must be shutdown for more than three months before this method can handle the decay heat load.

2. The plant needs to be in the refueling configuration; i.e., reactor head removed, reactor cavity flooded up and connected to the spent fuel pool. This permits the removal of the normal shutdown cooling system from operation and testing of these valves.

Only one additional non-refueling outage is scheduled during the current operating cycle. This is a three week surveillance / maintenance outage planned for spring 1994. The decay heat levels present during any outage less than several months precludes the use of the alternate cooling method without placing the plant in the refueling configuration.

The proposed change will preclude the need to place the plant in the refueling configuration prior to the next scheduled refueling outage. The level of Authority resources necessary for the removal of the drywell and reactor heads, and the connection of the reactor cavity to the spent fuel pool, is not practical solely for the purpose of testing the shutdown cooling isolation valves. Additionally, placing the plant in the refueling configuration will extend the length of the spring 1994 outage. Further, placing the plant in the refueling configuration to accommodate testing of the isolation valves will increase occupational radiation exposures.

III. SAFETY IMPLICATIONS OF THE PROPOSED CHANGE

The operating configuration of the shutdown cooling isolation valves and the RHR system when the reactor coolant system is pressurized (> 75 psig) substantially minimizes the possibility of gross leakage through these valves. A high reactor pressure interlock, as well as plant operating procedures, assures that these isolation valves are closed whenever the reactor pressure is above 75 psig. This protects the low pressure RHR system from overpressurization. The RHR system suction piping is designed for 450 psig. Gross leakage while the reactor is pressurized would be detected by high pressure on the RHR suction piping or an increase in suppression pool inventory. Consequently, the maintenance of normal operating status of the RHR system assures the absence of gross leakage through these valves.

These valves also receive an isolation signal in the event of a plant accident (reactor vessel low water level or high drywell pressure). This assures isolation of a potential leakage path from the reactor coolant system to the reactor building. For this path to exist, leakage through both isolation valves, and a breach of the RHR system piping would need to occur simultaneously. Since the isolation valves are maintained closed with the reactor pressurized, it is highly improbable that leakage through the valves will increase while the plant is operating. The redundant isolation valves provide two leakage barriers which limit the pathway leakage rate to that experienced by the valve with the smallest leakage rate. Further, the shutdown cooling system utilizes the RHR system whose entire pressure boundary is designed as a safety-related system. For these reasons, the potential for significant leakage to the reactor building by way of the shutdown cooling line is extremely remote.

The replacement of both isolation valves with valves of improved design provides added confidence that excessive leakage will not be experienced. The inboard valve (10MOV-18) was replaced during the 1985 refuel outage. This valve has experienced unacceptable leakage during only one of the four Type C tests performed during the refueling outages since its replacement. The outboard isolation valve (10MOV-17) was replaced with a new valve during the last refueling outage (1992). Considering the similarity in design of this valve to 10MOV-18, there is a high level of confidence in the overall integrity of the shutdown cooling penetration over the duration of the current operating cycle. The limited number of valve strokes these valves are subject to over any one operating cycle, minimizes valve degradation due to wear.

Based on the demonstrated reliability of the current shutdown cooling valve design, and the very low probability that the shutdown cooling system penetration would provide a pathway for significant leakage to the reactor building, for reasons described above, a deferral in the Type C tests for the shutdown cooling isolation valves does not pose a significant impact on plant safety.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the FitzPatrick plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change is limited to a one-time schedular extension in the shutdown cooling isolation valve Type C test. 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. The extension will not result in a significant increase in valve leakage considering that: (1) the valves are infrequently stroked, and then only when the reactor vessel is at low pressure, (2) monitoring the normal operating status of the RHR system assures the absence of gross valve leakage, and (3) the valves were replaced with valves of an improved design which has been confirmed by past Type C tests to exhibit satisfactory leak rate performance. For these reasons, the change does not involve a significant increase in the probability of an accident.

The change does not involve a significant increase in the consequences of an accident evaluated since any leakage through the shutdown cooling penetration will not significantly increase for reasons discussed in the previous paragraph, and such leakage is negligible compared to the main steam line break accident analyzed in the FSAR.

2. create the possibility of a new or different kind of accident from those 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 schedule extension for the shutdown cooling isolation valve Type C tests.

3. involve a significant reduction in the margin of safety.

   There is a very low probability of a significant increase in valve leakage considering the demonstrated reliability of the current valve design, the infrequent use of the valves, and the monitoring of the normal operating status of the RHR system. Moreover, any potential incremental benefit of performing the tests within the two year requirement would not be sufficient to offset the increased occupational radiation exposure associated with testing, and the risk to plant safety associated with the removal from service of the primary method of decay heat removal. Consequently, the proposed change does not involve a significant reduction in the margin of safety.

V. IMPLEMENTATION OF THE PROPOSED CHANGE

Implementation of the proposed changes will not affect the ALARA or Fire Protection Program at the FitzPatrick plant, nor will the change impact the environment.

VI. CONCLUSION

The change. as proposed, does not constitute an unreviewed safety question as defined in 10 CFR 50.59. That is, they:

1. will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report;

2. will not increase the possibility of an accident or malfunction of a type different from any previously evaluated in the Safety Analysis Report; and

3. will not reduce the margin of safety as defined in the basis for any technical specification.

The change therefore involves no signification hazards consideration, as defined in 10 CFR 50.92.

VII. REFERENCES

1. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report, Section 4.8.6.4, Shutdown Cooling Mode, and Section 7.3, Primary Containment and Reactor Vessel Isolation Control System.

2. NYPA letter, W. Fernandez to NRC (JAFP-90-742), October 11, 1990, "Reactor Containment Building Integrated Leakage Rate Test Report, James A. FitzPatrick Nuclear Power Plant."

3. NYPA letter, R. J. Converse to NRC (JAFP-87-534), April 30, 1987, "Reactor Containment Building Integrated Leakage Rate Test Report, James A. FitzPatrick Nuclear Power Plant."

4. NYPA letter, R. J. Converse to NRC (JAFP 85-0675), August 21, 1985, "Reactor Containment Building Integrated Leakage Rate Test Report, James A. FitzPatrick Nuclear Power Plant."