UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated January 11, 1994, the Power Authority of the State of New York (the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant, Technical Specifications (TS). Specifically, 10 CFR Part 50, Appendix J and TS 4.7.A.2.e(5) require the containment isolation valves to be leak rate tested during each refueling outage, but at intervals of not greater than two years. The licensee requested one-time relief from the requirement to perform Type C tests (local leak rate tests) at intervals of no greater than 2 years for the shutdown cooling isolation valves (10MOV-17 and 10MOV-18). This one-time only delay, until the next refueling outage currently scheduled to begin in November 1994, was requested for the performance of these leakage tests. The licensee's request was necessitated by the extended 1991-1993 refueling outage and the length of the current operating cycle.

2.0 BACKGROUND

The shutdown cooling isolation valves were previously leak rate 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 isolation valve 10MOV-17, and June 5, 1992 for the inboard isolation valve 10MOV-18. Subsequent delays in the outage resulted in these tests being performed significantly in advance of the start of the operating cycle (more than 7 months prior to the end of the outage). As a result, the 2-year test interval will be reached for these valves (May 30, 1994/June 5, 1994) 6 to 7 months prior to the next scheduled refueling outage. The amendment would permit a deferral in the performance of the Type C test of the shutdown cooling isolation valves beyond the 2-year limiting interval to the next refueling outage. Schedular exemption from the requirements of 10 CFR Part 50, Appendix J has been granted and is the subject of a separate letter.

The only effective means of removing reactor core decay heat is with the shutdown cooling mode of the RHR system. This requires both of the stated isolation valves to be in the open position. The shutdown cooling mode of the RHR system must be removed from service for approximately 24 hours to perform

9403250183 940318
PDR ADOCK 05000333
P PDR

a local leak rate test (Type C) of its isolation valves. 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 plant in the refueling condition. The alternate cooling me th the highest heat removal capacity is the Reactor Water Cleanup m in the blowdown mode. However, the reactor must be shutdown for more than 3 months before this method can handle the decay heat load.

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

A three week surveillance/maintenance outage is planned for spring 1994. However, 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 amendment would preclude the need to place the plant in the refueling configuration prior to the next scheduled refueling outage. Without the amendment, the licensee would be required to remove the drywell and reactor head d connect the reactor cavity to the spent fuel pool solely for the purpo f testing the shutdown cooling isolation valves. Placing the plant in the refueling configuration would significantly lengthen the spring 1994 outage and would require significant resources. Furthermore, placing the plant in the refueling configuration to accommodate testing of the isolation valves would significantly increase occupational radiation exposures. For these reasons, the licensee has determined that compliance with the TSs would result in undue hardship and costs.

3.0 EVALUATION

The operating configuration of the shutdown cooling isolation valves and the RHR system when the reactor coolant system is pressurized (greater than 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 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 improbable the 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 smallest leakage rate. For these reasons, the potential for significant leakage to the reactor building by way of the shutdown cooling line is minimal.

The penetration included in the licensee's schedular exemption request represents only 6.4 percent of the total "as left" leakage at the beginning of the current operating cycle. The total "as left" minimum path leakage for all penetrations was only 0.073 La and the total "as left" minimum path leakage for the penetration addressed in the proposed exemption was only 0.0046 La. 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 refueling outage and has successfully passed three out of four Type C tests performed during refueling outages since its replacement. The outboard isolation valve 10MOV-17 was replaced with a similarly designed new valve during the last refueling outage (1992). The limited number of valve strokes these valves are subject to over any one operating cycle minimizes valve degradation due to wear. This provides reasonable assurance that the requested surveillance interval expansion will not result in the Types B and C leakage rate total exceeding the 0.6 La limit of 10 CFR Part 50, Appendix J. Therefore, the NRC staff concludes that there are no significant radiological environmental impacts associated with the proposed schedular exemption.

The 2-year interval requirement for Type C testing is intended to be often enough to preclude significant deterioration between tests and long enough to permit the tests to be performed during routine plant outages. Leak rate testing of containment isolation valves during plant shutdown is preferable because of the lower radiation exposures to plant personnel. Furthermore, some containment isolation valves cannot be tested at power. For those valves that cannot be tested during power operation, or for which testing would yield unnecessary radiation exposure of personnel or involve unreasonable risk to personnel and equipment, the NRC staff believes the increase in confidence of containment integrity following a successful test is not significant enough to justify the hardships and costs associated with performing the tests within the 2-year time period. The licensee has presented information which gives a high degree of confidence that the components affected by this amendment will not degrade to an unacceptable extent. Acceptable leakage limits are defined in Sections III.B.3(a) and III.C.3 of Appendix J to 10 CFR Part 50. The NRC staff concludes that the potential incremental benefit of performing the tests within the 2-year 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 and extending the length of the spring 1994 surveillance/maintenance outage. Therefore, the NRC staff finds the amendment to be acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 4946). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
Brian C. McCabe

Date: March 18, 1994

March 18, 1994

Docket No. 50-333

Mr. William A. Josiger, Acting Executive
Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Josiger:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M88527)

The Commission has issued the enclosed Amendment No. 208 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated January 11, 1994.

The amendment provides one-time relief from the requirement to perform Type C tests (local leak rate tests) at intervals of no greater than 2 years for the shutdown cooling isolation valves (10MOV-17 and 10MOV-18). This one-time only delay, until the next refueling outage currently scheduled to begin in November 1994, was requested for the performance of these leakage tests. Your request was necessitated by the extended 1991-1993 refueling outage and the length of the current operating cycle.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,
Original signed by:
Brian C. McCabe, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 208 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDI-1:LA	PDI-1:PM	NRR/SCSB	OGC	PDI-1:D
NAME	CVogan	BMcCabe:avl	RBarrett	EHolcer	RACapra
DATE	2/17/94	02/17/94	3/7/94	3/14/94	3/18/94

OFFICIAL RECORD COPY
FILENAME: G:\FITZ\FIT88527.AMD