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March 27, 1992

Docket No. 50-423
B14086
Re: Operating License NPF-49

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 3
Changes to the Initial Test Program

The Millstone Unit No. 3 Operating License, NPF-49, contains License Condition 2.C.(10) which requires that "Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10CFR50.59 shall be reported in accordance with 50.59(b) within one month of such change." Northeast Nuclear Energy Company (NNECO) made a series of submittals in accordance with License Condition 2.C(10). Attachment 1, herein, provides a list of the submittals. In a letter dated May 10, 1989,(1) the NRC concluded that the changes to the Initial Test Program are acceptable and License Condition 2.C(10) had been met.

A recent engineering review of our initial test program revealed that NNECO failed to submit to the NRC one change to the Initial Test Program. Specifically, a test required for Item 71 of the Millstone Unit No. 3 Final Safety Analysis Report (FSAR) Table 14.2-1 was not performed during the initial start-up test program. At that time, Section 10 of Item 71 should have been deleted in accordance with 10CFR50.59 and notification to the NRC should have been made. On the contrary, this was not done and on February 24, 1992, this was identified by NNECO personnel and promptly reported to the NRC pursuant to Section 2.F of the Millstone Unit No. 3 Operating License, NPF-49 and subsequently a Licensee Event Report was filed on March 24, 1992 (Licensee Event Report 92-005). The Millstone Unit No. 3 start-up test program was developed based on the requirements of FSAR Chapter 14. A deficiency system documented any test or plant problems which occurred. Each deficiency was dispositioned by NNECO with respect to potential effects on plant operation and safety. We wish to emphasize that this was an isolated event in that all other changes to the Initial Test Program were reviewed against the provisions of 10CFR50.59 and reported to the NRC within 30 days as required by the Millstone Unit No. 3 License Condition 2.C(10).

(1) D. H. Jaffe letter to E. J. Mroczka, "Millstone Unit No. 3 Initial Test Program (TAC No. 60380)," dated May 10, 1989.

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The purpose of this letter is to formally document the above change to the Initial Test Program. Accordingly, NNECO hereby submits a report (Attachment 2) containing a brief description of the change to the Initial Test Program including a summary of the Safety Evaluation of that change. The indicated FSAR change will be included in a subsequent update to the FSAR.

If there are any questions, please contact our licensing representative directly.

Very truly yours,

FOR: J. F. Opeka
Executive Vice President

BY: W D Romberg
W. D. Romberg
Vice President

cc: T. T. Martin, Region I Administrator
V. L. Rooney, NRC Project Manager, Millstone Unit No. 3
W. J. Raymond, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

Attachment 1

Initial Test Program Tests changes Evaluated Under 10 CFR 50.59

	Reference NNECO Letter	Test (System)
1.	July 18, 1986	Functional (Spent Fuel Pool Cooling System)
2.	May 19, 1986	Functional (Boron Thermal Regeneration System)
3.	May 6, 1986	Functional (Main Feedwater System)
4.	May 2, 1986	Station Blackout (Turbine Generator)
5.	March 12, 1986	Reactor Trip Test, 50% Power (various)
6.	February 20, 1986	Hot, No Flow Rod Drop time (Control Rod Drive System)
7.	February 12, 1986	Functional, No Load, Operating Temperature and Pressure (Control Rod Drive Mechanism)

Attachment 2

Description of the Change:

The attached FSAR change deletes Item 10.a of Test 71 in FSAR Table 14.2-1. The item states that proper actuation, operation, reset and response time of the power operated relief valve (PORV) will be demonstrated by simulating a high pressure signal to each valve during a hot functional test (HFT).

Safety Evaluation

This test was not completed as stated. However, PORV 3RCS*PCV455A actually functioned satisfactorily to limit pressure during a plant trip on December 31, 1990 and IST 3-91-062 was recently performed to simulate a high pressure signal to PORV 3RCS*PCV456 and actuate a slave relay.

The proposed FSAR change was reviewed with respect to the requirements delineated in 10CFR50.59. It has been determined not to constitute an unreviewed safety question because it would not:

1. Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated.

   Millstone Unit No. 3 has two PORVs; 3RCS*PCV455A (PCV455A) and 3RCS*PCV456 (PCV456). They are designed to perform the following functions.

   - Prevent actuation of the reactor high pressure trip for all design transients.

   - Limit the RCS pressure excursion for some Anticipated Transient Without Trip (ATWS) events.

   - Provide cold overpressure protection (COPS) when the plant is shut down.

   Except for COPS, no credit is taken for operation of the PORVs in the safety analysis for design basis accidents. If the PORVs fail to open during a high pressure transient while the plant is at full power, the pressurizer safety valves will function to prevent RCS pressure from exceeding 110 percent of system design pressure in compliance with the ASME Code. The analysis for ATWS assumes both PORVs are available to function during some events. However, the effects of ATWS events are not considered as part of the design basis accident analysis for Millstone Unit No. 3.

The PORVs may be used for COPS. However, the circuitry and plant conditions applicable for COPS are different from the circuitry and plant conditions affected by the test. In any case, surveillance procedures are performed in order to assure COPS is operable.

Although the PORV test was not performed precisely as stated in the FSAR section 14, the start-up test program, the continuing surveillance test program, and actual PORV operation in response to a plant trip have verified that the PORVs are capable of performing their intended design function. Testing performed during start-up included functional checks of the PORV control circuits, verification that the PORV input signals were properly calibrated, and verification that the automatic controls operated in accordance with their design. PCV455A actually opened during a reactor/turbine overpressure event and prevented the pressurizer safety valves from lifting except for time response, this satisfied the requirements of the deleted test. Except for time response, the intent of the deleted test was also met for PCV456 through overlap testing during the HFT, including a manual discharge of steam to the Pressurizer Relief Tank.

A time response test was not performed for the complete operation of the PORVs during HFT. However, surveillance procedures are performed to determined time response for the pressure transmitters, and to verify that the PORVs will open within one second when manually operated from the control room. No time response testing is performed for circuitry between the pressure transmitters and the PORVs. But the time response of the intervening circuit is insignificant compared to the valve stroke time. In any event, there are no time response limits imposed by design. Therefore, the start-up and surveillance testing together with actual operation of a PORV when challenged have amply demonstrated that the PORV are capable of performing their safety and nonsafety-related design functions.

2. Increase the possibility for an accident or malfunction of a different type than any previously evaluated.

   No credit is taken for PORV operation during a transient as discussed above. Therefore, the failure to test high pressure actuation of a PORV during HFT will not create an accident of a different type than previously evaluated.

3. Decrease the margin of safety as defined in the basis for any Technical Specification.

   Since no credit is taken for operation of the PORV during a previously analyzed transient, there is no impact on the margin of safety.

TABLE 14.2-1 (Cont)

8. Demonstrate proper functioning of the main steam isolation valves at normal operating temperature and pressure.

9. Demonstrate the proper operation of steam generator safety valves, verifying setpoints with a pressure-assist device and verifying proper reseating and leakage within specified limits.

Q/A Question 8

10. Demonstrate the proper operation of pressurizer safety and relief valves, and the capability of the pressurizer relief tank to condense a steam discharge from the pressurizer.

~~Proper actuation, operation, reset and response time of the power operated relief valves (PORV) will be demonstrated by simulating a high pressure signal to each valve.~~ ~~640.19~~

a. ~~b.~~ The PORV will be operated manually to confirm valve operability and the ability of the pressurizer relief tank (PRT) to condense a discharge. Leakage following operation will be verified within acceptable limits. Discharge header leakage detection instrumentation will be verified operable in accordance with design requirements.

b. ~~c.~~ Operability of PORV and PRT instrumentation, controls, interlocks, and alarms will be verified.

c. ~~d.~~ Safety valve leakage at RCS normal pressure will be verified within specified limits. Actual safety valve operation will be demonstrated by hydrostatic bench test to verify set points.

11. Operate the reactor coolant pumps for a minimum of 240 hours at full flow to achieve approximately 1 million vibration cycles on reactor internals. Following hot functional testing, the internals are removed and inspected for vibration effects. See Section 3.9N.2.3 for additional information on the required inspection. 640.20 14

12. Demonstrate the operability of remote shutdown controls.

13. Perform or complete those portions of the following system tests (see individual descriptions), which require the RCS to be at or near normal operating temperature and pressure:

    a. Reactor coolant system expansion and restraint

    b. Chemical and volume control

    c. Boron thermal regeneration

    d. Residual heat removal