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SYSTEMATIC EVALUATION PROGRAM TOPIC VI-10.A, TESTING OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES, MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1

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NRC Research and Technical Assistance Report

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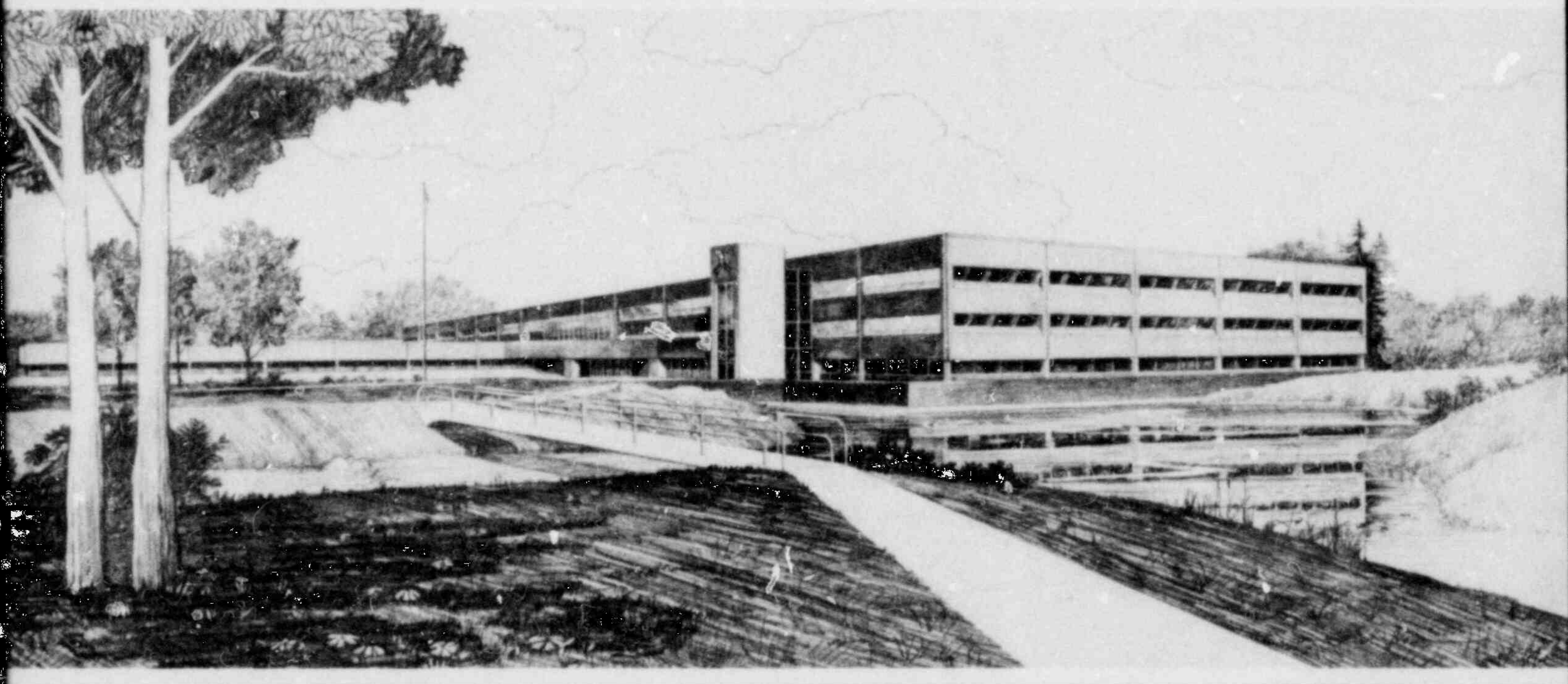

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SYSTEMATIC EVALUATION PROGRAM

TOPIC VI-10.A
TESTING OF REACTOR TRIP SYSTEM AND
ENGINEERED SAFETY FEATURES

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1

Docket No. 50-245

September 1981

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ABSTRACT

This SEP Technical Evaluation, for Unit Number 1 of the Millstone Nuclear Power Station, reviews the currently required component and system tests for the reactor trip system and for a typical engineered safety feature system. The currently required tests are then compared with current licensing criteria to determine if the required tests accomplish the same objectives as the licensing criteria.

FOREWORD

This report is supplied as part of the "Electrical, Instrumentation, and Control Systems Support for the Systematic Evaluation Program (II) being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Licensing by EG&G Idaho, Inc., Reliability & Statistics Branch.

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1.0 INTRODUCTION

The objective of this review is to determine if all Reactor Trip System (RTS) components, including pumps and valves, are included in component and system tests, if the scope and frequency of periodic testing is adequate, and if the test program meets current licensing criteria. The review will also address these same matters with respect to the Standby Liquid Control System (SLCS) as a typical example of all Engineered Safety Feature (ESF) systems.

2.0 CRITERIA

General Design Criterion 21 (GDC 21), "Protection System Reliability and Testability," states, in part, that:

> The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failure and losses of redundancy that may have occurred.[1]

Regulatory Guide 1.22, "Periodic Testing of the Protection System Actuation Functions," states, in Section D.1.a, that:

> The periodic tests should duplicate, as closely as practicable, the performance that is required of the actuation devices in the event of an accident;

and further, in Section D.4, states that:

> When actuated equipment is not tested during reactor operation, it should be shown that:

a. There is no practicable system design that would permit operation of the actuated equipment without adversely affecting the safety or operability of the plant,

b. The probability that the protection system will fail to initiate the operation of the actuated equipment is, and can be maintained, acceptably low without testing the actuated equipment during reactor operation, and

c. The actuated equipment can be routinely tested when the reactor is shut down.[2]

IEEE Standard 338-1977, "Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems," states, in part, in Section 3:

> Overlap testing consists of channel, train, or load-group verification by performing individual tests on the various components and subsystems of the channel, train, or load group. The individual component and subsystem tests shall check parts of adjacent subsystems, such that the entire channel, train or load group will be verified by testing of individual components or subsystems.[3]

and in part in Section 6.3.4:

> Response time testing shall be required only on safety systems or subsystems to verify that the response times are within the limits of the overall response times given in the Safety Analysis Report.
>
> Sufficient overlap shall be provided to verify overall system response.
>
> The response-time test shall include as much of each safety system, from sensor input to actuated equipment, as is practicable in a single test. Where the entire set of equipment from sensor to actuated equipment cannot be tested at once, verification of system response time shall be accomplished by measuring the response times of discrete

portions of the system and showing that the sum of the response times of all is within the limits of the overall system requirement.

In addition, the following criteria are applicable to the ESF: General Design Criterion 40 (GDC 40), "Testing of Containment Heat Removal System," states that:

> The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure:
>
> a. The structural and leaktight integrity of its components.
>
> b. The operability and performance of the active components of the system.
>
> c. The operability of the system as a whole and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection systems, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.[4]

GDC 38, "Testing of Emergency Core Cooling System," GDC 43, "Testing of Containment Atmosphere Cleanup Systems and GDC 46, "Testing of Cooling Water System," are similar.

Standard Review Plan, Section 7.3, Appendix A, "Use of IEEE Standard 279 in the Review of the ESFAS and Instrumentation and Controls of Essential Auxiliary Supporting Systems," states, in Section II.b, that:

> Periodic testing should duplicate, as closely as practical, the integrated performance required from the ESFAS, ESF systems, and their essential auxiliary supporting systems. If such a "system level" test can be performed only during shutdown, the testing done during power operation must be reviewed in detail. Check that "overlapping" tests do, in fact, overlap from one test segment to another. For example,

closing a circuit breaker with the manual breaker control switch may not be adequate to test the ability of the ESFAS to close the breaker.[6]

3.0 REACTOR TRIP SYSTEM (RTS)

3.1 Description. The system is made up of two independent logic channels, each having subchannels of tripping devices. Each subchannel has an input from at least one independent sensor, monitoring each of the critical parameters.

The output of each pair of subchannels is combined in a one-out-of-two logic: that is, an input in either one or both of the independent subchannels will produce a logic channel trip. Both of the other two subchannels are likewise combined in a one-out-of-two logic, independent of the first logic channel. The outputs of the two logic channels are combined in two-out-of-two arrangement so that they must be in agreement to initiate a scram. An off-limit signal in one of the two subchannels in one of the logic channels must be confirmed by any other off-limit signal in one of the two subchannels of the remaining logic channel to provide a reactor scram.

During normal operation, all vital sensor and trip contacts are closed, and all sensor relays are operated energized. The control rod pilot scram valve solenoids are energized, and instrument air pressure is applied to all scram valves. When a trip point is reached in any of the monitored parameters, a contact opens, de-energizing a relay which controls a contact in one of the two subchannels. The opening of a subchannel contact de-energizes a scram relay which opens a contact in the power supply to the pilot scram valve solenoids supplied by its logic channel. To this point, only one-half the events required to produce a reactor scram have occurred. Unless the pilot scram solenoids supplied by the other logic channel are de-energized, instrument air pressure will continue to act on the scram valves and operation can continue. Once a single channel trip is initiated, contacts in that scram relay circuit open and keep that circuit de-energized until the initiating parameter has returned within operating limits and the reset switch is actuated manually. It should be noted that each control rod has individual pilot scram solenoids for each channel and an individual air-operated scram valve. A normally-closed switch is provided in each logic channel pilot scram solenoid circuit. This allows each rod to be manually scrammed (tested) by opening both logic channel switches and de-energizing the pilot scram solenoids. This type of test would provide the required overlapping test of the RTS.

The parameters (sensors) which are required to initiate reactor scram are listed in Table 1. However, the only instruments included in this table are those required to prevent exceeding the fuel cladding integrity limits during normal operation or operational transients. These are described in Table VII-1 of the plant FSAR and listed in Tables 4.1.1 and 4.1.2 of the Millstone Nuclear Power Station Technical Specifications for Unit 1. For example, the condenser low-vacuum sensors are connected to the RPS trip system and can initiate a scram.

3.2 Evaluation. The Millstone 1 RTS is designed to allow overlapping tests from actuating device through the control rods. The design allows individual channel tests from sensors though pilot scram valves while the reactor is in operation and the overlapping rod scram tests during refueling. Although one or more rod scram valves may fail during reactor ope.. ion, the channel tests will verify that no common mode failure will occur and sufficient pilot valves will operate to shut down the reactor.

Table 1 shows the present Millstone 1 RTS instrument surveillance requirements, including frequency. The table also shows the current licensing requirements for General Electric boiling water reactors as listed in the Standard Technical Specifications. The tests shown only involve single channel testing (half-scram).

It should be noted that Technical Specification Table 4.1.2 does not require channel calibration for main steam-line isolation valve closure or turbine stop valve closure parameters, although the Millstone Technical Specification requirement for Unit 1 in Section 2.1.2.B requires that a 10%

TABLE 1. COMPARISONS OF MILLSTONE UNIT 1 RPS INSTRUMENT SURVEILLANCE WITH BWR STANDARD TECHNICAL SPECIFICATION REQUIREMENTS (STS)[6]

	Channel Check[a]		Channel Functional Test[b]		Channel Calibration[c]	
Instrument Channel	Millstone Unit 1	STS	Millstone Unit 1	STS	Millstone Unit 1	STS
High reactor pressure	NA	NA	Q*	M	Q	R
High drywell pressure	NA	NA	Q*	M	Q	Q
Low reactor water level	D	D	Q*	M	Q	R
High water level in scram discharge	NA	NA	Q*	M	Q	R
Condenser low vacuum	NA	NA	Q*[d]	NA	R	NA
Main steam-line isolation valve closure	NA	NA	Q*	M	NA	R
Turbine stop valves closure	NA	NA	Q*	M	M[e]	R
Manual scram	NA	NA	Q*	M	NA	NA
Turbine control valve fast closure	NA	NA	Q*	M	M[e]	Q
Average power range monitor (APRM) flow biased high flux	NA	S	Q*	SU[f]	W	W/SA
APRM-reduced high flux	NA	S	SU[f]	SU[f]	Q	W/SA
Intermediate range monitor (IRM)	D[e]	S	SU[f]	SU[f]	R	R
High steam line radiation	S	W	Q*	W	Q	R
Reactor mode switch in shutdown position	NA	NA	R	R	NA	NA

TABLE 1. (continued)

FREQUENCY NOTATION

Notation	Frequency	Notation	Frequency
S	At least once per 8 hours	R	At least once per refueling outage (18 months)
D	At least once per 24 hours	NA SA	Not applicable At least once per 184 days
W	At least once per 7 days	SU	Prior to start up
M	At least once per 31 days	SD	Prior to shutdown
Q	At least once per 3 months	Q*	Not less than one-month or greater than three months. Presently performed monthly.[10]

a. A qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel with other independent channels measuring the same variable.

b. Injection of a simulated signal into the channel to verify its proper response including, where applicable, alarm and/or trip initiating action.

c. Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip.

d. Consists of injecting a simulated electrical signal into the measurement channel.

e. This not required by technical specification, however, test are performed.[10]

f. Maximum test frequency is once per week.

valve closure initiate scram. These, and the time delay of 260 msec for the Turbine Control Valve Fast Closure are verified by surveillance procedures SP 408E, SP 408F and SP 408G resepctively, on a monthly basis.[10]

The Standard Technical Specifications for General Electric boiling water reactors (page 3/4 3-1, paragraph 4.3.1.2) require the logic system function test and simulated automatic operation at least every 18 months. This is done at Unit No. 1 of the Millstone Station by overlapping tests consisting of the half scram test and the scram insertion time test.

As can be seen in Table 1 the following channels are not subjected to a channel check as frequently as required for present-day licensing:

APRM--flow biased high flux
APRM--reduced high flux
IRM

The following channel is not subjected to a channel functional test as frequently as required for present-day licensing:

High steam line radiation

The following channels are presently given a channel functional test as frequently as required for present-day licensing; however, the technical specifications allow the present frequency to become quarterly, without notice to the NRC.

High reactor pressure
High drywell pressure
Low reactor water level
High water level in scram discharge
Main steam line isolation valve closure
Turbine stop valves closure
Manual scram
Turbine control valves fast closure
APRM--flow biased high flux

The following channel is not calibrated at least as frequently as required for present-day licensing:

APRM--reduced high flux

This should be a weekly calibration against heat balance calculations.

In Section 3.1 of the Millstone 1 Technical Specifications, 100 milliseconds is stated as the required limit to the response time between any channel trip and the de-energization of the scram solenoid relay. Response time testing to verify that the channel response time does not exceed this requirement is not in the technical specifications.

4.0 STANDBY LIQUID CONTROL SYSTEM

4.1 Description. The standby liquid control system is designed to insert a sodium pentaborate (or equivalent poison) solution to render and maintain the reactor subcritical even when the control rods are all fully withdrawn. The equipment consists of an unpressurized solution storage tank, a pair of positive displacement pumps, either of which has full capacity to perform the system function, two explosive actuated shear plug valves, a poison sparger ring and associated valves, piping and instrumentation. A complete description is in Section VI-7.2 of the plant FSAR.

The storage tank is heated to prevent particulate formation. The discharge of each pump is protected by a pressure relief valve that discharges back to the storage tank. Pilot light indication of circuit continuity for the explosive valves is provided. A single key controlled switch will start a pump and open associated valves. Both sets of valves and pumps are not operated simultaneously; however, the valves for both pumps may be open. A test tank and a supply of demineralized water are provided for testing.

The FSAR indicates that testing is done in two parts. One part determines the ability of the pump to develop flow and suction from the storage tank. The system is afterwards flushed to prevent boron precipitation. Another test uses demineralized water to show that water can be delivered

TABLE 2. STANDBY LIQUID CONTROL SYSTEM SURVEILLANCE REQUIREMENTS

Surveillance Requirements	Frequency: Millstone Unit 1	Frequency: STS
Solution temperature within limits.	D[a]	D
Solution volume is greater than specified.	D[b]	D
Heat traced pump suction piping is greater than or equal to 70°F.	D[a]	D
Start both pumps and recirculate demineralized water to the test tank.	M[c]	M
Verify the continuity of the explosive charges.	D[d]	M
Solution chemical analysis.	M/M	M/M
Verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.	M[d]	M
Initiating one loop using demineralized water and replacement of the explosive charge.	R	R
Verify minimum flow requirement against reactor vessel head pressure.	M	R
Demonstrate relief valve setpoint and that it does not operate during recirculation test to the test tank.	R[e]	R
Verify piping from the storage tank to the reactor vessel is not blocked.	R/M[d]	R/M
Demonstrate that the storage tank heaters are operable.	a	R

a. Minimum temperature is 75° per surveillance procedure SP 641.2. This provides an indirect test of the operability of the storage tank heaters.

b. Minimum volume is specified by technical specification Figures 3.4.1 and 3.4.2.

c. Flow rate required to be 32 gpm while the FSAR design requires 40 gpm. The technical specifications do not require testing of both pump loops; surveillance procedure SP 661.4 does. Pressure is not specified. Technical specification, 4.4.A.2b recirculates solution from and to the storage tank at least once in 18 months for both systems.

d. Not in technical specifications, required by surveillance procedure.[10]

e. Non-operation during recirculation test is not required. Surveillance procedure SP 662.1 verifies that the relief valves do not operate under normal system operating pressure.

into the reactor vessel. This test requires replacement of the explosive charges in the shear plug valves.

4.2 Evaluation. Table 2 shows the current testing requirements for the standby liquid control system and associated systems. The surveillance required by technical specifications and surveillance procedures is done at least as frequently as required for present day licensing.

The Millstone 1 technical specifications do not agree with the design presented in the FSAR, in that the minimum test flow rate is 80% of the design flow rate.

Further, it is apparent that Millstone 1 has only one three-phase heater in the solution storage tank, whereas present requirements are for two redundant heaters.

5.0 SUMMARY

The Technical Specifications for Millstone Unit 1 were compared with the Standard Technical Specifications for current Boiling Water Reactor licensing. It was found that, for the reactor trip system, three signals are not subjected to a channel check, one signal is not subjected to a channel functional test and one channel is not calibrated as frequently as required in the standard technical specifications. (See Section 3.2.) Additionally, the channel response time between channel trip and the de-energization of the scram relay is not required to be tested.

For the Standby Liquid Control System, selected as typical of ESF systems, surveillance requirements were as frequent as required in the standard technical specifications.

6.0 REFERENCES

1. General Design Criterion 21, "Protection System Reliability and Testability," of Appendix A, "General Design Criteria of Nuclear Power Plants," 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

2. Regulatory Guide 1.22, "Periodic Testing of the Protection System Actuation Functions."

3. IEEE Standard 338-1975, "Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems."

4. General Design Criterion 40, "Testing of Containment Heat Removal Systems," of Appendix A, "General Design Criteria of Nuclear Power Plants," 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facil ties."

5. Nuclear Regulatory Commission Standard Review Plan, Section 7.3, Appendix A, "Use of IEEE Standard 279 in the Review of the ESFAS and Instrumentation and Controls of Essential Auxiliary Supporting Systems."

6. Standard Technical Specifications for General Electric Boiling Water Reactors (BWRs), NUREG-0123, Revision 2, Fall 1980.

7. Millstone Point Nuclear Power Station-Unit No. 1, "Final Safety Analysis Report," Amendment 5, dated March 14, 1968.

8. Technical Specifications and Bases for Millstone Nuclear Power Plant Unit 1, Appendix A, to Provisional Operating License DPR-21, Amendments 1 through 45, dated December 1977.

9. Northeast Utilities letter, W. G. Counsil to Director of Nuclear Reactor Regulation, NRC, "SEP Topic VI-10.A, Testing of Reactor Trip System and Engineered Safety Features," August 4, 1981, A01766.