NORTHEAST UTILITIES

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HOLYOKE WATER POWER COMPANY
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August 25, 1986

Docket No. 50-423
B12201

Office of Nuclear Reactor Regulation
Attn: Mr. V. S. Noonan, Director
PWR Project Directorate #5
Division of PWR Licensing - A
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Reference: (1) W. G. Counsil letter to B. J. Youngblood, three (N-1) loop operation, dated November 20, 1984.

Dear Mr. Noonan:

Millstone Nuclear Power Station, Unit No. 3
Three (N-1) Loop Operation

In Reference (1), Northeast Nuclear Energy Company (NNECO) submitted the design basis information for the modification to the solid state protection system such that the system can be configured for three-loop operation. A meeting was held between NRC Staff and NNECO representatives in Bethesda, Maryland on July 28, 1986 to discuss the Staff's concerns regarding the aspects of the protection systems which are unique to plant operation with a loop out of service. A discussion of reactor protection system interlocks, setpoints, alarms/indicators, single failure criterion, operator actions, and on-line testing with the plant in three-loop operation was provided to the Staff at the meeting. The Staff indicated that their concerns were satisfactorily resolved. However, the Staff requested that NNECO provide the proposed FSAR changes and certain detailed drawings related to the reactor coolant system loop isolation valve interlocks. Accordingly, the information contained in Enclosure I has been prepared and is being forwarded directly to Ms. E. L. Doolittle, the NRC Project Manager for Millstone Unit No. 3. Appropriate changes to the FSAR pages will be forwarded to the NRC in a future FSAR update.

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

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka
Senior Vice President

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ENCLOSURE I

Item 1: The Reactor Coolant System (RCS) loop isolation valve interlocks

The attached FSAR pages 7.2-19, 7.2-20 and 7.6-4 and FSAR Figure 7.2-1 (Sheets 17, 18, 19) provide information regarding the reactor trip system instrumentation trip setpoints and the RCS loop isolation valve interlocks. These FSAR pages and figures will be included in a future FSAR update.

Item 2: Final Design Drawings

A copy for each of the drawings listed in Attachment I was provided to the staff at the July 28, 1986 meeting. This list will be incorporated into FSAR Section 1.7 in a future FSAR update.

Attachment I

Title	Drawing No.
Two train protection system	1083H88 Sheets 1-32
Window Arrangement - Main Control Board MB-4	ESK-10BL
Window Arrangement - Main Control Board MB-4	ESK-10BN
Annunciator Input List Reactor First-out Alarm	ESK-10KC
Annunciator Input List Reactor Trip Bistable Status-Lights	ESK-10KF
Loop Stop Valve Protection Cabinet Sheets 1-4	8759D83
Elem. Diag. 480V 3RCS*8001A Reac. Clnt LP1 Hot Leg Stop VV	ESK-6TF
Elem. Diag. 480V 3RCS*8002A Reac. Clnt LP1 Cold Leg Stop VV	ESK-6TK
Reac. Clnt LP Stop VV AUX Isolation Circuit	ESK-7RB
Reac. Clnt LP Stop VV AUX Isolation Circuit	ESK-7RC

Proposed FSAR Changes

Bypass and inoperable alarms are in accordance with Regulatory Guide 1.47. |2

INSERT (A) →

7.6.5 Reactor Coolant System Loop Isolation Valve Interlocks

The purpose of these interlocks is to ensure that an accidental startup of an unborated and/or cold, isolated reactor coolant loop results only in a relatively slow reactivity insertion rate.

The interlocks allow opening of the cold leg loop stop valves ~~are required to perform a protective function. Therefore, there are:~~ (refer to Valve 2 on Fig 7.6-4) whenever:

1. ~~Two independent limit switches to indicate that a valve is fully open~~ The hot leg isolation valve (Valve 1 on Fig 7.6-4) is opened, and
2. ~~Two independent limit switches to indicate that a valve is fully closed~~ Loop temperatures are less than a preset amount (170°F), and
3. ~~Two differential pressure switches in each line which bypasses a cold leg loop isolation valve. This is the line which contains the relief line isolation (Valve 4 on Figure 7.6-4). It should be noted that flow through the relief line isolation valves indicates: 1. the valves in the line are open, 2. the line is not blocked, and 3. the pump is running.~~

INSERT (B) → Temperature differences among loops are less than 20°F.

7.6.6 Fuel Pool Cooling and Purification System

7.6.6.1 Description

The fuel pool cooling and purification system design is described in Section 9.1.3, and the flow diagram is shown on Figure 9.1-6.

Fuel pool cooling pump motor controls are located on the main control board and at the switchgear. REMOTE/LOCAL control selector switches are provided at the switchgear. An annunciator is alarmed on the main control board when local control is selected. One fuel pool cooling pump and one fuel pool purification pump are in continuous service when spent fuel is in the fuel pool. Should low water level in the fuel pool be alarmed, the operator can stop the operating fuel pool cooling pump and the operating fuel pool purification pump and take corrective steps to locate and isolate the leak in the system.

The following parameters are indicated on the fuel pool cooling panel:

1. Fuel pool water level
2. Fuel pool demineralizer total flow
3. Fuel pool water temperature
4. Fuel pool coolers outlet temperature

INSERT A

Startup of an isolated reactor coolant loop is prevented by strict administrative control until the plant is in Mode 5 or 6 with all conditions of Technical Specification 3/4.4.1.6 satisfied.

INSERT B

For the logic functions of these interlocks refer to to Figure 7.2-1 Sheets 17,18, and 19.

DNBR is not a directly measurable quantity; however, the process variables that determine DNB are sensed and evaluated. Small isolated changes in various process variables may not individually result in violation of a core safety limit; whereas the combined variations, over sufficient time, may cause the overpower or overtemperature safety limit to be exceeded. The design concept of the reactor trip system takes cognizance of this situation by providing reactor trips associated with individual process variables in addition to the overpower/overtemperature safety limit trips. Process variable trips prevent reactor operation whenever a change in the monitored value is such that a core or system safety limit is in danger of being exceeded should operation continue. Basically, the high pressure, low pressure and overpower/overtemperature ΔT trips provide sufficient protection for slow transients as opposed to such trips as low flow or high flux which will trip the reactor for rapid changes in flow or flux, respectively, that would result in fuel damage before actuation of the slower responding ΔT trips could be affected.

Therefore, the reactor trip system has been designed to provide protection for fuel cladding and reactor coolant system pressure boundary integrity where:

1. A rapid change in a single variable of factor which will quickly result in exceeding a core or a system safety limit

2. A slow change in one or more variables will have an integrated effect which will cause safety limits to be exceeded.

   Overall, the reactor trip system offers diverse and comprehensive protection against fuel cladding failure and/or loss of reactor coolant system integrity for Condition II and III accidents. This is demonstrated by Table 7.2-4 which lists the various trips of the reactor trip system, the corresponding technical specification on safety limits and safety system settings and the appropriate accident discussed in the safety analyses in which the trip could be utilized.

It should be noted that the reactor trip system (RTS) ~~automatically~~ using prescribed administrative procedures provides core protection during non-standard operating configuration, i.e., operation with a loop out of service. Although operating with a loop out of service over an extended time is considered to be an unlikely event, no protection system setpoints need to be reset. This is because the nominal value of the power (P-8) interlock setpoint restricts the power such that DNBRs less than 1.30 will not be realized during any Condition II transients occurring during this mode of operation. This restricted power is considerably below the boundary of permissable values as defined by the core safety limits for operation with a loop out of service. Thus the P-8 interlock acts essentially as a high nuclear power reactor trip when operating with one loop not in service. By first resetting the coefficient setpoints in the overtemperature ΔT function to more restrictive

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values as listed in the technical specifications, the P-8 setpoint can then be increased to the maximum value consistent with maintaining DNBR above 1.30 for Condition II transients in the one loop shutdown mode. The resetting ~~of the ΔT Overtemperature trip and P-8~~ will be carried out under prescribed administrative procedures, under the direction of authorized supervision, and with the plant conditions prescribed in Section 3.4.1.1 of the Technical Specifications.

of the reactor trip system instrumentation trip setpoints as listed in the Technical Specifications

The RTS design meets the requirements of Criterion 21 of the 1971 GDC.

Preoperational testing is performed on reactor trip system components and systems to determine equipment readiness for startup. This testing serves as a further evaluation of the system design.

Analyses of the results of Condition I, II, III, and IV events, including considerations of instrumentation installed to mitigate their consequences, are presented in Chapter 15. The instrumentation installed to mitigate the consequences of load rejection and turbine trip is given in Section 7.4.

7.2.2.2.2 Reactor Coolant Flow Measurement

The elbow taps used on each loop in the primary coolant system are instrument devices that indicate the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow has occurred. The correlation between flow and elbow tap signal is given by the following equation:

$$\frac{\Delta P}{\Delta P_o} = \left(\frac{w}{w_o}\right)^2, \qquad (7.2\text{-}3)$$

Where ΔP_o is the pressure differential at the reference flow w_o, and ΔP is the pressure differential at the corresponding flow, w. The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The expected absolute accuracy of the channel is within ±10 percent of full full and field results have shown the repeatability of the trip point to be within ±1 percent.

7.2.2.2.3 Evaluation of Compliance to Applicable Codes and Standards

The reactor trip system meets the criteria of the general design criteria as indicated. The reactor trip system meets the requirements of Section 4 of IEEE Standard 279-1971, as indicated below: