Docket No. 50-245

Attachment No. 1

Millstone Nuclear Power Station, Unit No. 1

Additional Information Regarding
Reload 8/Cycle 9 License Amendment Submittal

November, 1982

7. Scram Discharge System

In Reference (1), we identified the setpoint for the scram discharge volume high water level trip in Table 3.1.1 as being less than or equal to 24 inches above the centerline of the lower end cap to the scram discharge instrument volume (SDIV) pipe weld. To reflect the as-built configuration, a setpoint of less than or equal to 26 inches is now necessary. The high water level trip precludes from occurring the situation whereby the scram discharge volume (SDV) fills with water to the extent that water discharged to the SDV from a reactor trip could not be accomodated and thus would result in slow scram times or partial or no control rod insertion. The high water level trip setpoint is determined such that 3.34 gallons per control rod drive still remains available in the SDV. This criterion was developed by the BWR Owners' Group Ad-Hoc Committee on I & E Bulletin No. 80-17 and concurred with by the NRC Staff in their Generic Safety Evaluation Report, dated December 9, 1980.

Similarly, the setpoint for scram discharge volume high water level to initiate control rod block needs to be changed from 12 inches above lower cap to the SDIV pipe weld to 14 inches. The basis for this setpoint is that a rod block would occur at a level which is approximately one-half the scram high water level. This is consistent with our previous 39-gallon and 18-gallon high water level setpoints for reactor scram and control rod block, respectively.

A technical review of these revised setpoints has found them to be acceptable. Also, a safety evaluation has been performed in accordance with 10CFR50.59 and has concluded that these changes do not constitute any unreviewed safety questions. The Millstone Nuclear Review Board has reviewed and approved the proposed changes and has concurred with the above determination.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum Number of Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in which Function Must Be Operable			Action*
			Refuel (8)	Startup/Hot Standby	Run	
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM:					
3	High Flux	≤ 120/125 of full scale	X	X	(5)	A
3	Inoperative	A. HI Voltage < 80 volt DC B. IRM Module Unplugged C. Selector Switch not in Operate Position	X	X	X (10)	A
	APRM:					
2	Flow Biased High Flux	See Section 2.1.2A	X	X	X	A or B
2	Reduced High Flux	See Section 2.1.2A	X	X	X	A or B
2	Inoperable	A: > 50% LPRM Inputs** B. Circuit Board Removed C. Selector Switch not in Operate Position	X	X	X	A or B
2	High Reactor Pressure	≤ 1085 psig	X	X	X	A
2	High Drywell Pressure	≤ 2 psig	X (9)	X (7)	X (7)	A
2	Reactor Low Water Level	≥ 1.0 inch***	X	X	X	A
2	Scram Discharge Vol. High Level	≤ 26 inches above the center-line of the lower end cap to SDIV pipe weld	X (2)	X	X	A

TABLE 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable Instrument Channels per Trip System(1)	Instrument	Trip Level Setting
2	APRM Upscale (Flow Biased)	See Specification 2.1.2B
2	APRM Downscale	$\geq$ 3/125 Full Scale
1 (6)	Rod Block Monitor Upscale (Flow Biased)	$\leq$ .65 w + 42 (2)
1 (6)	Rod Block Monitor Downscale	$\geq$ 3/125 Full Scale
3	IRM Downscale (3)	$\geq$ 3/125 Full Scale
3	IRM Upscale	$\leq$ 108/125 Full Scale
2	SRM Detector not in Startup Position	(4)
2 (5)	SRM Upscale	$\leq 10^5$ counts/sec.
1	Scram Discharge Volume - Water Level High	$\leq$ 14 inches above lower cap to SDIV pipe weld
1	Scram Discharge Volume - Scram Trip Bypassed	N/A

(1) For the Startup/Hot Standby and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks; IRM downscale are not operable in the RUN position and APRM downscale need not be operable in the Startup/Hot Standby mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

(2) W is the total core flow in percent of design (69×10^6 #/hr.). Trip level setting is in percent of full power.

(3) IRM downscale may be bypassed when it is on its lowest range.

(4) This function may be bypassed when the count rate is $\geq$ 100 cps or when all IRM range switches are above Position 2.

(5) One of these trips may be bypassed. The SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

APRM's #4, #5 and #5 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water, generator load rejection, and turbine stop valve closure are discussed in Section 2 of these specifications.

Instrumentation (pressure switches) in the drywell is provided to detect a loss of coolant accident and initiate the emergency core cooling equipment. This instrumentation is a backup to the water level instrumenatation which is discussed in Specification 3.2. A scram is provided at the same setting as the emergency core cooling system (ECCS) initiation to minimize the energy which must be accommodated during a loss of coolant accident and to prevent the reactor from going critical following the accident.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this piping is an instrument volume which is the low point in the piping. No credit was taken for the volume contained in the piping below a point which is 26 inches above the lower cap to the SDIV pipe weld when calculating the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrumented volume which alarm and scram the reactor while there is still greater than 3.34 gallons per drive available to accept scram water. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient with bypass closure. Ref. Section 4.4.3 FSAR. The condenser low vacuum scram is a back-up to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 23" Hg vacuum, stop valve closure occurs at 20" Hg vacuum and bypass closure at 7" Hg vacuum.

Docket No. 50-245

Attachment No. 2

Millstone Nuclear Power Station, Unit No. 1

Additional Information Regarding
Reload 8/Cycle 9 License Amendment Submittal

November, 1982

1. Is the increased MCPR due to the code or increased fuel loading?

The increased MCPR is strictly attributed to the use of ODYN. The increased fuel loading is offset by Gadolinia and by design would have been the same or less than last cycle.

2. In Cycle 8 the MCPR (measured) was close to 1.50. In Cycle 9 how close will it come?

In Cycle 9 the predicted MCPR is as follows:

MWD/ST	MCPR	RAPLHGR
0	1.63	0.83
200	1.63	0.81
1000	1.63	0.79
2000	1.65	0.80
2500	1.64/1.61	0.82/0.78
3000	1.59	0.80
4000	1.62	0.85
5000	1.66/1.61	0.85/0.85
6000	1.66	0.70
7000	1.50	0.77
7500	1.59	0.76
8000	1.68	0.73
8540	1.77	0.67

NOTE: The MCPR at 7000 MWD/ST is 1.50. Review of the proposed rod pattern at that point and the MCPR's around that point show it to be out of line. A value of 1.60 to 1.65 would be planned for by adjusting the rod pattern.

3. In comparing the axials in the GESTAR they are different than the ELLA axials.

The axials shown in the ELLA report were generated for plant "H" and are shown as an illustration of the bottom peaking effect of the low flow regime. These axials are shown as a demonstration not as a bounding set of axials.

4. The ELLA report does not appear to have a low flow case for Millstone. The Cycle 9 reverification appears to have a low flow case.

True. The case for the low flow rod block intercept is based on the ELLA report which demonstrates that for both BWR-4 and BWR-3 type plants that the 100/100 case is always more limiting.

The reverification was performed for the upcoming cycle and cases were run at 100/100 and at 100/87 as stated. The runs were made with ODYN and were made utilizing the limiting MCPR event, which is the load reject without bypass. The intent was to demonstrate that the 100/100 case was still more limiting than the 100/87 event.

This intent was met as seen in the heat flux for the event.

100/100	Q/A	126.53 %
100/87	Q/A	125.89 %

5. Regarding high burnup MAPLHGR limits previously imposed by the NRC Staff due to fission gas concerns, verify that the positions stated in R.E. Engel (GE) letters to T. A. Ippolito (NRC) on "Extension of ECCS Performance Limits," dated May 6, 1981 and May 28, 1981, are applicable to Millstone Unit No. 1.

   The above-mentioned references have been reviewed and are both applicable to Millstone Unit No. 1. Specifically, no additional credit is taken in our analyses for the ECCS evaluation model changes and the peak cladding temperature for each fuel type and burnup does not exceed that assumed in Table 2 of the May 28, 1981 letter.

6. Verify that the composition of the Segmented Test Rod (STR) bundle at Millstone Unit No. 1 is consistent with what was expected in light of the discrepancies found by General Electric after the 1980-1981 refueling outage.

   The STR bundle at Millstone Unit No. 1 has been verified to consist of fuel segments as described in our May 20, 1981 letter (#B10210) to D. M. Crutchfield and NEDE-20592-5P, Revision 1, "STR Bundle Submittal Millstone 1 Segmented Test Rod Bundle Supplement 5," dated May 1981, which was submitted to the NRC Staff by our May 20, 1981 letter (#B10211). Therefore, the conclusions contained in our May 20, 1981 letter (#B10210) are still valid. GE has informed us that a report describing the current status of the STR bundle at Millstone Unit No. 1 will be prepared and transmitted to us. Subsequent to the receipt of this report, a submittal will be made to the NRC Staff.

7. If the turbine by-pass function is credited in your feedwater controller failure to maximum demand transient analysis, commit to propose Technical Specification changes to require surveillance testing of the turbine by-pass valves.

   The turbine by-pass function is credited in our feedwater controller failure to maximum demand transient analysis. However, we are not certain how many of the ten (10) by-pass valves are required to operate such that this transient is not the limiting MCPR or vessel pressure transient. We intend to request General Electric to determine this transient's sensitivity to turbine by-pass valves' operability for Millstone Unit No. 1. It is our understanding that GE can perform this work in approximately two to three months. The sensitivity to bypass valve availability will be utilized to determine the degree of surveillance that would be appropriate for Millstone Unit No. 1. During this process, we intend to consider that:

(i) the pressure sensing and hydraulic systems for the turbine by-pass valves are both common to the control valves which are continually verified operable,

(ii) the turbine by-pass and control valves are mechanically linked together, and can be visually inspected,

(iii) operation of any by-pass valves increases ALARA concerns with respect to the condenser,

(iv) a 10% power loss is incurred everytime one by-pass value is opened.