Docket No. 50-245
B11663

Attachment No. 1

Proposed Technical Specification Changes
for
Millstone Unit No. 1, Reload 10

August, 1985

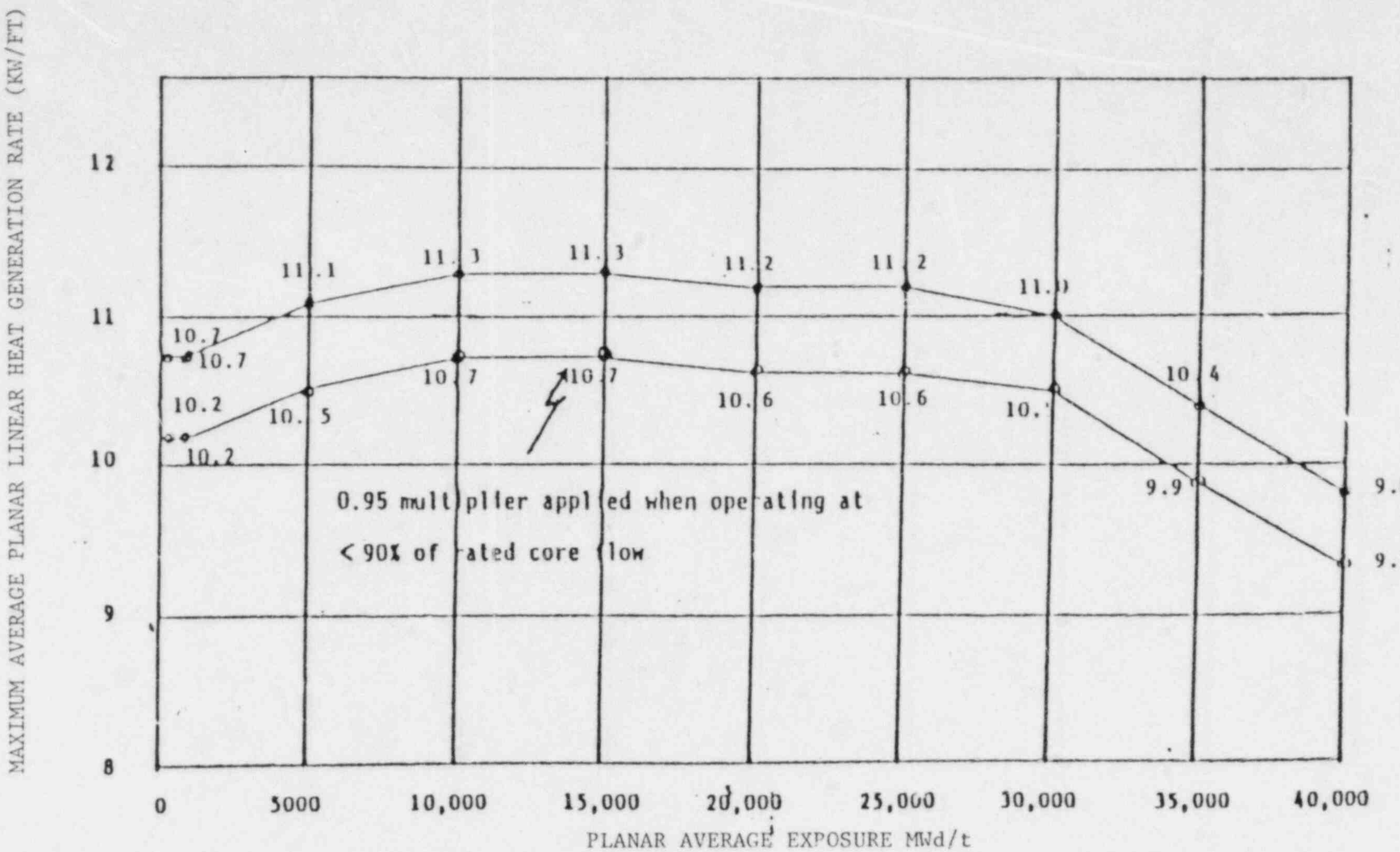

Figure 3.11.1a - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE P8DRB282

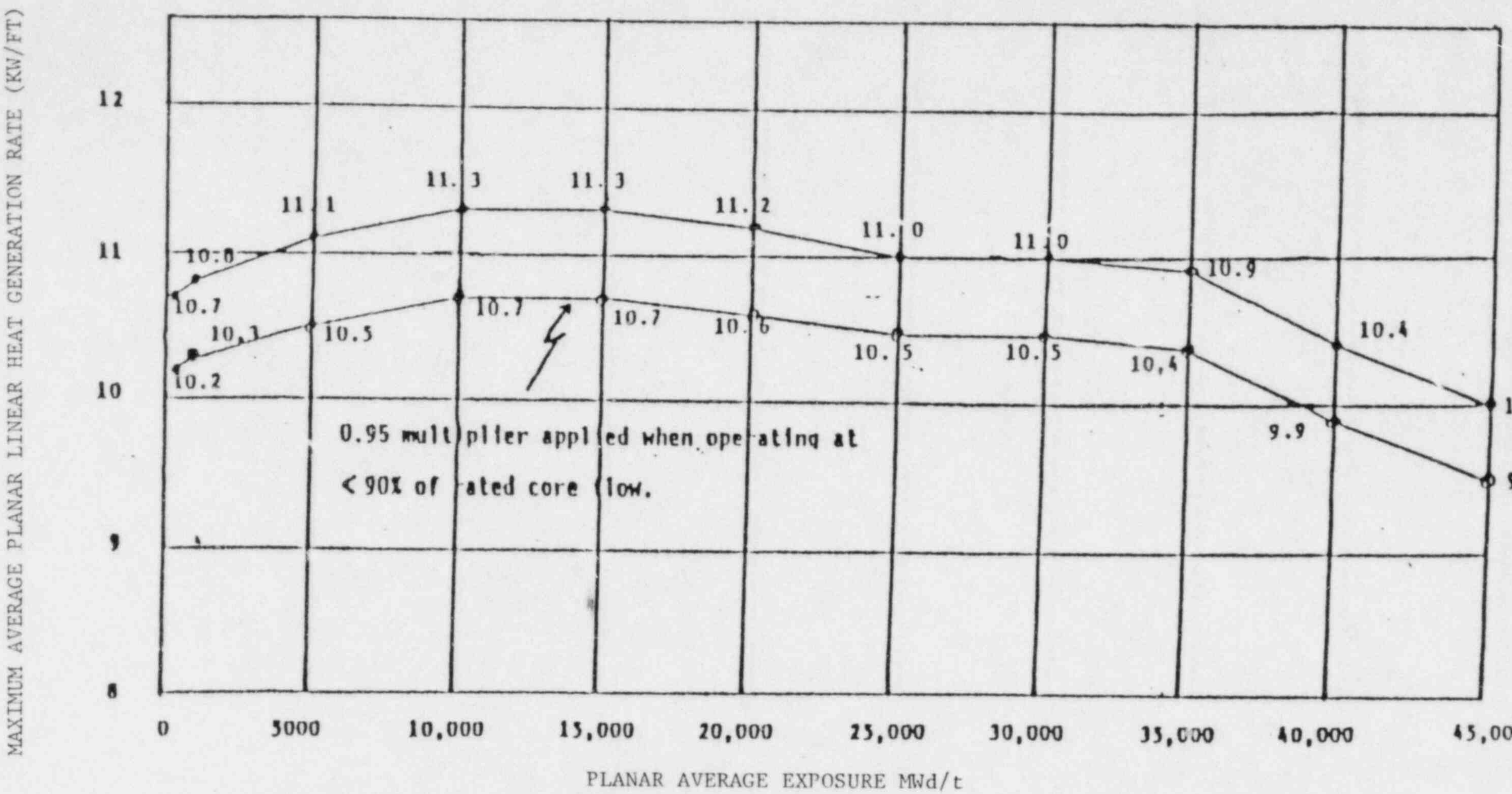

Figure 3.11.1b - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE P8DRB283H

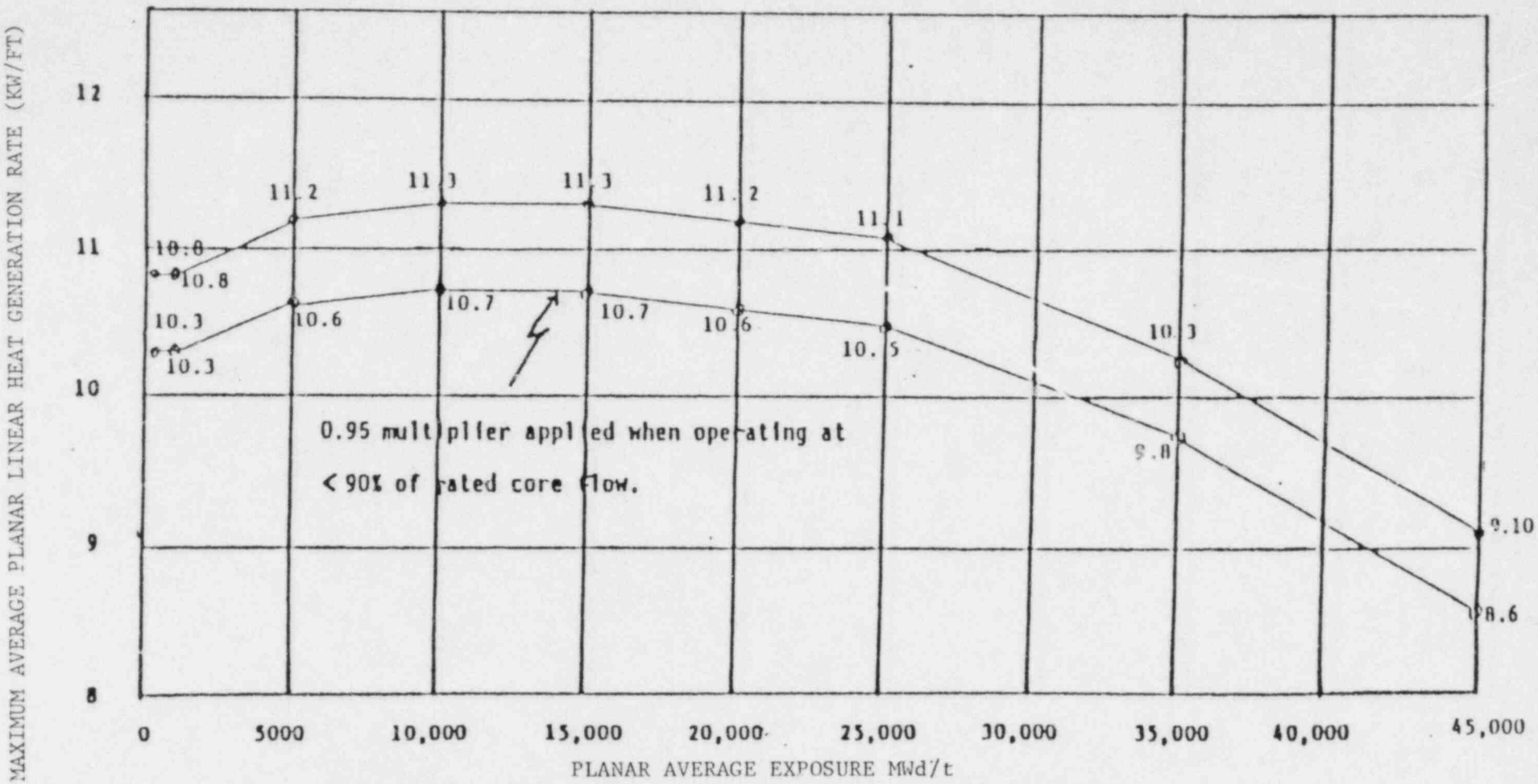

Figure 3.11.1c - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE BP8DRB300

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TABLE 3.11.1

OPERATING LIMIT MCPR'S FOR CYCLE 10

(OPTION B)

BOC 10 TO EOC 10	EOC 10 TO 70% COASTDOWN	FUEL TYPE
1.42	1.42	P8 x 8R
1.42	1.42	BP8 x 8R

OPERATING LIMIT MCPR'S FOR CYCLE 10

(OPTION A)

BOC 10 TO EOC 10	EOC 10 TO 70% COASTDOWN	FUEL TYPE
1.48	1.48	P8 x 8R
1.48	1.48	BP8 x 8R

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate a reactor scram and provide automatic isolation of the Reactor Protection System buses from their power supplies.

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1.

B. Response Time

The time from initiation of any channel trip to the de-energization of the scram solenoid relay shall not exceed 50 milliseconds.

C. Reactor Protection System Power Monitoring

Two RPS electric power monitoring channels for each inservice RPS MG set or alternate power supply shall be operable at all times except as follows:

1. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to OPERABLE status within 72 hours or remove the associated RPS MG set or alternate power supply from service.
2. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to OPERABLE status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram and provide automatic isolation of reactor protection system buses from their power supplies.

Objective:

To specify the type and frequency of surveillance to be applied to the reactor protection instrumentation.

Specification:

A. Instrumentation system shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

B. Daily during reactor power operation, the maximum fraction of limiting power density shall be checked and the APRM scram and rod block settings given by the evaluations in Specifications 2.1.2A and 2.1.2B shall be determined to be valid.

C. The RPS electrical protection assemblies shall be determined operable as follows:

1. At least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST, and
2. At least once per 18 months by demonstrating the OPERABILITY of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers, and verifying the following setpoints:
   a. Over-voltage $\leq$ (132)VAC,
   b. Under-voltage $\geq$ (108) VAC,
   c. Under-frequency $\geq$ (57)Hz, and
   d. Time-delay $\leq$ (4.0) seconds.

TABLE 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable Instrument Channels per Trip System(1)	Instrument	Trip Level Setting
1(7)	APRM Upscale (Flow Biased)	See Specification 2.1.2B
1(7)	APRM Downscale	$\geq$ 3/125 Full Scale
1 (6)	Rod Block Monitor Upscale (Flow Biased)	$\leq$ .65W + 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 recirculation flow required to achieve rated core flow expressed in percent.

(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.

power operation. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collect housing.

B. Control Rod Withdrawal

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. However, for purposes of removal of a control rod drive, as many as one drive in each quadrant may be uncoupled from its control rod so long as the reactor is in the shutdown or refuel condition and Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number of inoperable rods has been reduced to less than three and if it has been demonstrated that control rod drive mechanism collet housing failure is not the cause of an immovable control rod.

B. Control Rod Withdrawal

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

   a. when the rod is fully withdrawn the first time subsequent to each refueling outage or after maintenance, observe that the drive does not go to the overtravel position; and

   b. when the rod is withdrawn the first time subsequent to each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation; however, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is critical shall be performed to verify instrumentation response.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection shall be recorded.

D. Coolant Leakage

Any time irradiated fuel is in the reactor vessel, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 2.5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, initiate an orderly shutdown and have the reactor in the cold shutdown condition within 24 hours.

E. Safety and Relief Valves

1. During power operation or whenever the reactor coolant pressure is greater than 90 psig with irradiated fuel in the reactor vessel, the safety valve function of the six relief/safety valves shall be operable, except as specified in 3.6.E.5 below. (The solenoid activated relief function of the relief/safety valves shall be operable as required by Specification 3.5.D.).

2. If Specification 3.6.E.1 is not met, initiate an orderly shutdown and have the reactor coolant pressure below 90 psig within 24 hours.

3. When the safety/relief valves are required to be operable per Specification 3.6.E.1, the Valve Position Indication shall be operable. Two of the six channels may be out of service provided backup indication for the affected valves is provided by the Valve Discharge Temperature Monitor.

D. Coolant Leakage

Reactor coolant system leakage into the primary containment shall be checked and recorded at least once per day.

E. Safety and Relief Valves

1. Three of the relief/safety valves top works shall be bench checked or replaced with a bench checked top works each refueling outage. All six valves top works shall be checked or replaced every two refueling outages. The set pressure shall be adjusted to correspond with a steam set pressure of:

No. of Valves	Set Point (psig)
1	1095 ± 1%
1	1110 ± 1%
4	1125 ± 1%

2. At least one of the relief/safety valves shall be disassembled and inspected each refueling outage.

3. During each operating cycle with the reactor at low pressure, each safety valve shall be manually opened until operability has been verified by torus water level instrumentation, or by the Valve Position Indication System, or by an audible discharge detected by an individual located outside the torus in the vicinity of each discharge.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
2. If Specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped. Operation with a single recirculation pump is permitted for 24 hours unless the recirculation pump is sooner made operable. If the pump cannot be made operable, the reactor shall be in cold shutdown within 24 hours. 3. The reactor shall not be operated unless the equalizer line is isolated. 4. With the mode switch in the startup/hot standby or run mode, operation without forced circulation shall not be permitted.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
5. All station and switchyard 24 and 125 volt batteries and associated battery chargers are operable.	c. During the monthly generator test, the diesel fuel oil transfer pumps shall be operated.
B. When the mode switch is in Run, the availability of power shall be as specified in 3.9.A, except as specified below:	2. Gas Turbine Generator
1. From and after the date that incoming power is available from only one 345 kv line, reactor operation is permissible only during the succeeding seven days unless an additional 345 kv line is sooner placed in service.	a. The gas turbine generator shall be fast started and the output breakers closed within 48 seconds once a month to demonstrate operational readiness. The test shall continue until the gas turbine and generator are at equilibrium temperature at full load output. Use of this unit to supply power to the system electrical network shall constitute an acceptable demonstration of operability.
2. From and after the date that incoming power is not available from any 345 kv line, reactor operation shall be permitted provided both emergency power sources are operating and the isolation condenser system is operable. The NRC shall be notified, within 24 hours of the precautions to be taken during this situation and the plans for restoration of incoming power. The minimum fuel supply for the gas turbine during this situation shall be maintained above 20,000 gallons.	b. During each refueling outage, the conditions under which the gas turbine-generator is required will be simulated and a test conducted to verify that it will start and be able to accept emergency loads within 48 seconds.
3. From and after the date that either emergency power source or its associated bus is made or found to be inoperable for any reason, reactor operation is permissible according to Specification 3.5.F/4.5F unless such emergency power source and its bus are sooner made operable, provided that during such time two offsite lines (345 or 27.6 kv) are operable.	B. Batteries 1. Station Batteries a. Every week the specific gravity and voltage of the pilot cell and temperature of adjacent cells and overall battery voltage shall be measured. b. Every three months the measurements shall be made of voltage of each cell to nearest 0.01 volt, specific gravity of each cell and temperature of every fifth cell.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
B. Core Monitoring	B. Core Monitoring

B. Core Monitoring

During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant, except as specified in Paragraphs 3 and 4 below. For an SRM to be considered operable, the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal operating level. (Use of special movable dunking type detectors during fuel loading or major core alterations in place of normal detectors are permissible as long as the detector is connected into the normal SRM circuit.)

2. The SRM shall have a minimum neutron induced count rate of three per second with all rods fully inserted in the core.

3. Prior to unloading, the SRM's shall be proven operable as stated above, however, during spiral unloading, the count rate may drop below 3 cps.

4. If required, special movable dunking type detectors can be inserted into the core, prior to reloading fuel assemblies into the central core region (with all control rods inserted). Before the ninth fuel assembly is loaded into the core in the close proximity of the movable dunking chambers or the SRM;s Paragraph 3.10.B.1 and 2 apply.

SURVEILLANCE REQUIREMENT

B. Core Monitoring

1. Prior to making any alterations to the core, the SRM's shall be functionally tested and checked for neutron response. Thereafter, the SRM's will be checked daily for response when core alterations are being made.

2. Prior to spiral unloading or reloading, the SRM's shall be functionally tested. Prior to spiral unloading, the SRM's should also be checked for neutron response.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds seven times normal background. The purpose of the scram is to reduce the source of such radiation to the extent necessary to prevent excessive release of radioactive materials. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation of the main condenser off-gas line provided the limit for a 15-minute period specified in specification 3.8 is not exceeded.**

The main steamline isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open in three out of four lines. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting the resultant transient is insignificant. Ref. Section 11.3.7 FSAR.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. Section 7.2 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM and APRM system provide protection against excessive power levels and short reactor periods in the refuel and Startup/Hot Standby modes. A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions. Thus the IRM and APRM systems are required in the refuel and Startup/Hot Standby modes. In the power range the APRM provides the required protections; thus, the IRM system is not required in the Run mode.

The high reactor pressure, high drywell pressure, reactor low water level, and scram discharge volume high level scrams are required for Startup/Hot Standby and Run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have all scram functions except those listed in Note 8 of Table 3.1.1 operable in the Refuel and Shutdown mode is to assure that shifting to the Refuel mode during reactor power operation does not diminish the need for the reactor protection system. As indicated in Note 11 of Table 3.1.1, no trip functions are required to be operable if all control rods are fully inserted, valved out and electrically disarmed, since this condition assures maximum negative reactivity insertion.

** Per errata sheet dated 10-7-70

3.2 Bases:

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitagate the consequences of accidents which are beyond the operator's ability to control or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are to assure the effectiveness of the protective instrumentation when required by its capability to tolerate a signal failure of any component of such systems even during periods when portions of such systems are out of service for maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations and to prescribe the trip settings required to assure adequate performance.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in table 3.2.1 which senses the conditions for which isolation is required. Such instrumentaion must be available whenever primary containment integrity is required.The objective is to isolate the primary containment so that the guideline values of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127 inches above the top of the active fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps. Ref. Section VII-4.4 FSAR. For a trip setting of 127 inches above the top of the active fuel and a 60-second valve closure time the valve will be closed before core uncovery occurs even for the maximum break in the line; and therefore, the setting is adequate.

The low low reactor water level instrumentation is set to trip when reactor water level is 79 inches above the top of the active fuel. This trip initiates closure of Group 1 primary containment isolation valves, Ref. Section VII-4.4 FSAR and also activates the ECC subsystems and starts the emergency diesel generator and the gas turbine generator and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that post accident cooling can be effectively accomplished and the guideline values of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria.The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Ref. Section VI-2.7 FSAR. The Isolation Condenser system has been added to the ECC system to insure that cladding integrity is maintained for postulated small break LOCA conditions in the recirc. discharge piping with a gas turbine failure and LPCI injection into the damaged loop.

(1) NEDO-24085-1, Loss-of-Coolant Accident Analysis Report for Millstone Unit 1 Nuclear Power Station.

High pressure actuation of the Isolation Condenser (IC) will be a backup to direct activation on Low-Low level; similar to other ECCS systems. Activation is based on the high pressure signal (1085 PSIG for 15 seconds) which occurs after MSIV closure on Low-Low water level, SRV actuation, and subsequent repressurization. The activation of the IC requires only the opening of normally closed valve IC-3 in the condensate return line. This valve is powered by the safety-grade DC battery. All valves in the system are powered by safety-grade AC or DC power and are also used for containment isolation. All are normally in the open position (other than IC-3). The IC system is safety Class 2 and is seismically qualified. The shell side water volume is sufficient for approximately 30 minutes of operation at rated conditions without makeup. Two sources of makeup are available. For small break mitigation, less than 10 minutes of operation is required, and generally at less than rated conditions.

Two sensors on the isolation condenser supply and return lines are provided to detect line failure and actuate isolation action. The sensors on the supply and return sides are arranged in a 1 out of 2 logic and to meet the single failure criteria, all sensors and instrumentation are required to be operable. The isolation settings and valve closure times are such as to prevent core uncovery or exceeding site limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to < 1.07. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, eight IRM's, or four SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the IRM and RBM may be reduced by one for a short period of time to allow for maintenance testing and calibration.

The APRM rod block trip is flow biased and prevents significant approach to MCPR=1.07 especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that fuel damage limits are not exceeded.

The RBM provides local protection of the core, i.e., the prevention of fuel damage in a local region of the core, for a single rod withdrawal error. The trip point is flow biased. The worst case single control rod withdrawal error has been analyzed for the initial core and also prior to each reload; the results show that with specified trip settings, rod withdrawal is blocked within an adequate margin to fuel damage limits. This margin varies slightly from reload to reload and, thus, each reload submittal contains an update of the analysis. Below ∿ 70% power, the withdrawal of single control rod results in MCPR > 1.07 without rod block action, thus requiring the RBM system to be operable above 30% of rated power is conservative. Requiring at least half of the normal LPRM inputs from each level to be operable assures that the RBM response will be adequate to prevent rod withdrawal errors.

The IRM rod block functions assure proper upranging of the IRM system, and reduce the probability of spurious scrams during startup operations.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough or the neutron flux is below the instrument response threshold. In these cases the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale trips are set at 3/125 of full scale.

4.2 Bases:

The instrumentation listed in Table 4.2.1 will be functionally tested and calibrated at regularly scheduled intervals. Although this instrumentation is not generally considered to be as important to plant safety as the Reactor Protection System, the same design reliability goal of 0.99999 is generally applied for all applications of (1 out of 2) X (2) logic. Therefore, on-off sensors are tested once/3 months, and bi-stable trips associated with analog sensors and amplifiers are tested once/week.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1 out of n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel.[2] This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = \sqrt{\frac{2t}{r}}$$

where i = the optimum interval between tests
t = the time the trip contacts are disabled from performing their function while the test is in progress
r = the expected failure rate of the relays.

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 minutes to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hour. Using this data and the above operation, the optimum test interval is:

$$i = \sqrt{\frac{2(0.5)}{10^{-6}}} = 1 \times 10^{3} \text{ hours} = 40 \text{ days}$$

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure operability of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

The above calculated test interval optimizes each individual channel, considering it to be independent of all others. As an example, assume that there are two channels with an individual technician assigned to each. Each technician

(2) UCRL-50451 Improving Availability and Readiness of Field Equipment Through Periodic Inspection, Benjamin Epstein, Albert Shiff, July 16, 1968, Pg. 10, Equation (24), Lawrence Radiation Laboratory.

3. The peak fuel enthalpy content of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in reference 1.

Since Millstone Unit No. 1 has referenced the report, General Electric Standard Applications for Reload Fuel (Reference 4), the assumptions regarding the control Rod Drop Accident are applicable to Millstone Unit No. 1. By using the analytical models described in this report coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 20% of rated power, the specified limit on in-sequence control rod or control rod segment worths will limit the peak fuel enthalpy content to less than 280 cal/gm. Above 20% power even single operator errors cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalpy content of 280 cal/gm should a postulated control rod drop accident occur.

Each core reload will be analyzed to show conformance to the follwoing bounding conditions:

a. Accident reactivity curves equal to or less than those assumed in Reference (4).
b. Doppler reactivity coefficients equal to or more negative than those assumed in Reference (4).
c. Up to $0.02\Delta k$, actual scram reactivity feedback function equal to or greater than data presented in Reference (4).

If the above conditions are all met, then the reload is within the generic RDA analysis. If any one of these conditions is not met, then a more detailed, plant-specific evaluation will have to be performed to demonstrate compliance with the design limit of 280 cal/gm.

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(3) Stirn, R. C., Paone, C. J., and Haun, J. M., "Rod Drop Accident Analysis of Large Boiling Water Reactor Addendum No. 2 Exposed Cores," Supplement 2 - NEDO-10527, January 1975.

(4) NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel.

It is recognized that these bounds are conservative with respect to expected operating conditions. If any one of the above conditions is not satisfied, a more detailed calculation will be done to show compliance with the 280 cal/gm design limit.

Should a control rod drop accident result in a peak fuel energy content of 280 cal/gm, less than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in offsite doses twice that previously reported in the FSAR, but still well below the guideline values of 10 CFR 100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate, which has nearly the same consequences as for the 7 x 7 fuel case because of the operating rod power differences.

The RWM provides automatic supervision to assure that out-of-squence control rods will not be withdrawn or inserted: i.e., it limits operator deviations from planned withdrawal sequences. References Section VII.10 of the FSAR. It serves as an independent backup of the normal withdrawal procedure followed by the operator. In the event that the RWM is out of service when required, a second independent operator or engineer can manually fulfill the operator follower control rod pattern conformance function of the RWM. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 20% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is required to be operating during a startup for the withdrawal of a significant number of control rods for any startup.

The scram times for all control rods will be determined at the time of each refueling outage. The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system and also verifies the ability of the control rod drive to scram since if a rod can be moved with drive pressure, it will scram because of higher pressure applied during scram . The frequency of exercising the control rods under the conditions of three or more control rods out of service provides even further assurance of the reliability of the remaining control rods.

The occurrence of scram times within the limits, but significantly longer than average, will be viewed as a possible warning of systematic problem with control rod drives especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

D. Control Rod Accumulators

The specification for the number of accumulators which may be valved out of service is based on a series of two dimensional XY diffusion theory calculations at 20°C. These analyses prove that the reactor will be subcritical even when the central control rod of each 3 x 3 nine rod array is fully withdrawn.

E. Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operating base cconditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% Δk are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

F. Power/Flow Operating Map

Allowable combinations of thermal power and total core flow are restricted to Curve 1 of Figure 3.3.1. Analyses show that reactor ascension to full power may proceed on a modified power/flow line bounded by the rod block line up to a point called the 100% intercept point (100% power/87% total core flow), from which continued power increases may proceed in a direct linear manner to the 100% power/100% flow point (5,6,7).

(5) "Millstone Point Nuclear Power Station - Unit 1 Load Analysis License Amendment Submittal"NEDO-21285, June,1976.

(6) "Millstone Unit 1 - Load Line Limit Analysis" NEDO-21285-1, November,1977.

(7) "Extended Load Line Limit Analysis - Millstone Point Nuclear Power Station Unit 1" ,NEDO-24366,September,1981.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, hot standby and reactor startup. During these periods with steaming rates less than 1 percent of full flow (80,000 pounds per hour), a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations are maintained at conservative levels. At steaming rates of at least 1 percent of full flow (80,000 pounds per hour), boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When conductivity becomes abnormal then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt; e.g., Na_2SO_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWRs, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include operation of the reactor cleanup system, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the cleanup system to reestablish the purity of the reactor coolant. During startup periods and hot standby, which are in the category of less than 1% of full flow (80,000 pounds per hour), conductivity may exceed 2 mho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 mho (other than short-term spikes), samples will be taken to assure that the chloride concentration is less than 0.1 ppm.

The conductivity at the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. While conductivity monitoring assures that pH is in the normal range, samples of reactor coolant are taken and tested for pH once a week as a check. Isotopic analyses to determine major contributors to activity can be performed by a gamma scan.

D. Coolant Leakage

The 2.5 gpm limit for leaks from unidentified sources was established by assuming the leakage was from the primary system. Tests demonstrate that a relationship exists between the size of a crack and the probability that a crack will propagate.

A. 1. Primary Containment

The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site doses to values less than those specified in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.5% ΔK. A drop of a 1.5%ΔK rod does not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well within 10 CFR 100 guideline values.

2. Suppression Chamber

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system or for releases through the safety relief valves. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig.

Since all of the gases in the drywell are considered purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 42 psig which is below the design of 62 psig. Maximum water volume of 100,400 ft^3 results in a downcomer submergence of 3.33 feet and, the minimum volume 98,000 ft^3 results in a submergence of 3.0 feet. The majority of the Bodega tests were run with a submerged length of four feet and with complete condensation. Additional condensation tests were run in the Mark I Full Scale Test Facility (FSTF) at downcomer submergence varying between 1.5 and 4.5 feet and complete condensation of steam resulted. Thus, with respect to downcomer submergence, this specification is adequate.

The maintenance of a drywell-suppression chamber differential pressure of 1.00 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.0 to 3.33 feet will assure the post-LOCA suppression pool swell hydrodynamic forces are minimized and consistent with loads assumed for structural analysis of the suppression chamber.

5. Oxygen Concentration

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent of so) reaction of the zirconium and steam during a loss of coolant accident would lead to the liberation of sufficient hydrogen to result in a flammable concentration in the containment. Subsequent ignition of the hydrogen if it is present in sufficient quantities to result in excessively rapid recombination, could result in a loss of containment intergrity.

The 4% oxygen concentration minimizes the possibility of hydrogen combustion following a loss of coolant accident. Significant quantities of hydrogen could be generated if the core cooling systems did not sufficiently cool the core.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is more probable than the occurrence of the loss of coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety without significantly reducing the margin of safety. Thus to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24 hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration. The primary containment is normally slightly pressurized during periods of reactor operation assuring no air in-leakage through the primary containment. However, at least once a week, the oxygen concentration will be determined as added assurance.

B. Standby Gas Treatment System

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. Both standby gas treatment system fans are designed to automatically start upon containment isolation and to maintain the reactor building pressure to the design negative pressure so that all leakage should be in-leakage. Each of the two fans has 100 percent capacity.

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal absorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine absorbers. The charcoal absorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal absorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a

radioactive methyl iodide removal efficiency of at least 95 percent for expected accident conditions. If the efficiencies of the HEPA filters and charcoal absorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HDPA filters and charcoal absorbers.

Only one of the two standby gas treatment systems is needed to clean up the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. During refueling two off-site power sources (345KV or 27KV) and one emergency power source would provide an adequate and reliable source of power and allow completion of annual diesel or gas turbine preventative maintenance.

C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required.

D. Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

5.0 DESIGN FEATURES

5.1 Site

The Unit 1 reactor building is located on the site at Millstone Point in Waterford,Connecticut.The nearest site boundary on land is 1620 feet northeast of the reactor building, which is the minimum distance to the boundary of the exclusion area as described in 10 CFR 100.3(a). No part of the site which is closer to the reactor building than 1620 feet shall be sold or leased except to (i) The Connecticut Light and Power Company, Western Massachusetts Electric Company or Northeast Nuclear Energy Company or their corporate affiliates for use in conjunction with normal utility operations and (ii) to the two leasees under the leases referred to in the following paragraph.

A United States Navy research Laboratory and a desalination pilot operation of the Maximum Evaporator Division of the Cuno Engineering Corporation may be permitted to operate within the exclusion area under leases which make activities and persons on the leased premises subject to health and safety requirements of the owner of the site.

5.2 Reactor

A. The core shall consist of 580 fuel assemblies.

B. The reactor core shall contain 145 cruciform-shaped control rods. The control material shall be hafnium and/or boron carbide powder (B_4C) compacted to approximately 70% of theoretical density.

5.3 Reactor Vessel

The reactor vessel shall be as described in Table IV-1 of the FSAR. The applicable design codes shall be as described in Table IV-1 of the FSAR.

5.4 Conatainment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table V-1 of the FSAR.

B. The secondary containment shall be as described in Section V-3 of the FSAR and the applicable codes shall be as described in section XII of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set fourth in Section V-2 of the FSAR.

5.5 Fuel Storage

A. The new fuel starage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.90.

5.6 Seismic Design

The reactor building and all contained engineered safeguards are designed for maximum credible earthquake ground motion with an acceleration of 17% of gravity. Dynamic analysis was used to determine the earthquake acceleration applicable to the various elevations in the reactor building.

Docket No. 50-245
B11663

Attachment No. 3

Millstone Unit No. 1 Technical Specification
Typographical Error Corrections, Clarifications
and Reference Updates

August,1985

1. T.S. 4.1.B-Reactor Protection System, Page 3/4 1-1

   Change "peak heat flux" to "maximum fraction of limiting power density" to match present day terminology. Also, the words "to be valid" are added for clarification.

2. TABLE 3.2.3 - INSTRUMENTATION THAT INITIATES ROD BLOCK, Page 3/4 2-5

   Rod Block Monitor (RBM) upscale equation was incorrectly changed in Table 3.2.3 in a prior submittal. This change corrects that error

3. T.S. 3.3.B.1 - REACTIVITY CONTROL, CONTROL ROD WITHDRAWAL, Page 3/4 3-2

   Removal of the word "not" from the phrase "as many as one drive in each quadrant may be uncoupled from its control rod so long as the reactor is not in the shutdown or refuel condition" is required for clarification. Control rod drives are not removed for maintenance purposes at power as required by the wording of the existing Specification. Dose, temperature, shutdown margin, and control rod drive housing support restraints dictate that rod drive removal only occur in the shutdown or refuel condition. Therefore,the removal of the word "not" is required.

4. T.S. 3.6.F - SAFETY AND RELIEF VALVES, Page 3/4 6-5

   a. Change "F" to "E" in this section title to provide correct lettering sequence.

   b. Remove the word "Acoustic" from 4.6.E Surveillance Requirements. A previous Technical Specification Change removed the word from the Limiting Condition For Operation section on the same page.

5. T.S. 3.6.H.3 - RECIRCULATION PUMP FLOW MISMATCH, Page 3/4 6-11.

   The terms 'equalizer valves are closed" and "equalizer line is isolated" are equivalent. In either case the recirculation loops are isolated from each other.

6. T.S. 3.9.B.2, AUXILIARY ELECTRICAL SYSTEM, Page 3/4 9-2.

   Change the word "the" to "and".

7. T.S. 3.10.B.4 - REFUELING AND SPENT FUEL HANDLING, CORE MONITORING, Page 3/4 10-2.

   This Technical Specification is being rewritten to allow flexibility in core monitoring during fuel loading. If 3 counts per second (cps) can be maintained on the in-core

Source Range Monitors (SRMs) by loading irradiated fuel near them early in the fuel loading sequence, dunking chambers will not be required. The proposed wording change provides this flexibility.

8. 3.1 Bases, Page B 3/4 1-3

Reference to Table 3.1.1 indicates the note should be NOTE 8 rather than NOTE 7 as stated.

9. 3.2 BASES

a. Page B 3/4 2-1. The existing FSAR reference is incorrect. Also the Bases are updated to reference the current LOCA analysis.

b. Page B 3/4 2-2a. This change provides a clarification of the sequence of events following MSIV closure for actuation of the Isolation Condenser on high reactor pressure.

c. Page B 3/4 2-3. The words "of 127 inches of water and 79 inches of water" are not trip settings of the Isolation Condenser and are being removed.

10. 4.2 BASES, Page B 3/4 2-5

The reference number is being changed from (1) to (2) to maintain consistency of numbering of references in the Bases.

11. 3.3.B BASES, CONTROL ROD WITHDRAWAL

a. Page B 3/4 3-3. The Bases are updated to reflect current General Electric Rod Drop Accident methodology and References.

b. Page B 3/4 3-4. An incorrect section of the FSAR is referenced. The section should be V11.10 rather than 7-9.

12. 3.3.F BASES, POWER/FLOW OPERATING MAP, Page B 3/4 3-6.

The Bases are updated to reflect current power/flow map and restrictions, including references to Load Line Limit Analysis and Extended Load Line Limit Analysis.

13. 3.6 and 4.6 BASES, COOLANT CHEMISTRY, Page B 3/4 6-3

Review of this paragraph indicates that the word "normal" should actually be "abnormal".

14. 3.7 BASES

a. Page B 3/4 7-1. The section title is added.

b. Page B 3/4 7-5 and 7-6. The Titles and Sections are rearranged to provide consistency and order in this section of the Bases.

15. 5.0 DESIGN FEATURES

a. 5.2 REACTOR, Page 5-1. This section is updated to include the provision for control rods containing hafnium.

b. 5.5 FUEL STORAGE, Page 5-2. Section 5.5.B requires K_{eff} of the Spent Fuel Pool to be less than or equal to 0.90. The criteria in Section 5.5.C that U235 loading be less than or equal to 15.2 gm/cm was derived without taking credit for Gadolinia or reactivity depletion due to burnup. These overly conservative assumptions result in maximum allowable calculated bundle enrichments lower than that of General Electric production bundles currently being used in most BWR's. The removal of Section 5.5.C is justified as long as the condition specified in Section 5.5.B is satisfied at all times.

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Attachment No.2

"Supplemental Licensing Submittal for
Millstone Unit No. 1 Reload 10"
23A4696, dated August 1985

August,1985