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NORTHEAST NUCLEAR ENERGY COMPANY

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March 5, 1979

Docket No. 50-245

Director of Nuclear Reactor Regulation
Attn: Mr. D. L. Ziemann, Chief
Operating Reactors Branch #2
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 1
Provisional Operating License No. DPR-21
Reload #6 License Amendment Submittal

Pursuant to 10CFR50.90, the holders of Provisional Operating License, DPR-21, hereby propose a change in the next refueled core as described in the attachment, Reload 6, Supplemental Reload Licensing Submittal.

The reload fuel consists of 148 General Electric 8x8R bundles, with average bundle enrichment of 2.65 w/o, identical to fuel type 8DBR265H&L described in "GE/BWR Generic Reload Fuel Application", NEDE-24011-P-A, Revision 0, August, 1978. All 148 bundles will have drilled lower tie plates and finger springs to regulate bypass flow. Fuel channels of an improved design based on developmental channels will be installed on all new reload fuel. The channels are similar to those installed during the past few refuelings except for a special heat-treatment which is intended to reduce channel corrosion.

The Technical Specification changes proposed in the attachment to this letter are derived from the reanalysis of certain limiting transients and accidents for the reloaded core configuration as described in the enclosed submittal.

The submittal contains a discussion of operation in a coastdown mode beyond the end of full power life. As the analysis shows, end-of-cycle limits are conservative operating conditions for this mode; and, therefore, no new specifications are required for coastdown to 70% power. An additional analysis reported in the attached supplement demonstrates the ability to operate at end of cycle with a reduction in feedwater temperature. Thermal limits at the end of cycle conservatively bound operation in this mode.

Also included in this submittal is NEDE-20592-4P, "STR Bundle Submittal, Millstone Unit 1 Segmented Test Rod Bundle", Supplement 4, January, 1979. This supplement updates the original report, submitted October 3, 1974, as part of the Reload No. 2 license amendment submittal, which was supplemented July, 1975, August, 1976, and November, 1977, and describes proposed changes to be made in the bundle composition during the spring, 1979, refueling outage. We have been informed by the General Electric Company that this information is considered proprietary by them, and as such, it is being forwarded under separate cover.

The present schedule calls for plant shutdown on April 21, 1979. A five-week outage is anticipated.

The Nuclear Review Board has reviewed and approved the Technical Specification changes for this Reload No. 6 License Amendment Submittal.

NNECO has reviewed the above proposed License Amendment pursuant to the requirements of 10CFR170, and has determined that the proposal constitutes a Class IV amendment. Accordingly, enclosed herewith is payment in the amount of $12,300 (twelve thousand three hundred dollars). The basis for this determination is that the proposal involves several changes of the Class III type.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W G Counsil

W. G. Counsil
Vice President

Attachments

Submittal includes three signed and notarized originals and 37 copies.

STATE OF CONNECTICUT)
) ss. Berlin March 5, 1979
COUNTY OF HARTFORD)

Then personally appeared before me W. G. Counsil, who being duly sworn, did state that he is Vice President of Northeast Nuclear Energy Company, a Licensee herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Licensees herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Sheila M. Oates
Notary Public

My Commission Expires March 31, 1981

DESCRIPTION OF TECHNICAL SPECIFICATION CHANGES

THERMAL LIMITS

Maximum Average Planar Linear Heat Generation Rate and Minimum Critical Power Ratios,

Section	Page
Figure 3.11.1h	3/4 11-7b
Table 3.11.1	3/4 11-10
Bases 2.1.1	B2-1, B2-2
Bases 2.1.2A, B, E, F	B2-6, B2-7, B2-8
Bases 3.2	B3/4 2-3

The revised operating limit MAPLHGR's and CPR's are the result of transient analysis performed with the Cycle 7 core. The limits are supported by the attached topical reports NEDO-24168 and NEDO-24168-1, Supplemental Reload Licensing Submittals for Millstone 1, Reload 6.

A technical review of these specifications has found them to be acceptable; a safety evaluation has also 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.

CONTROL ROD WITHDRAWAL

Control Rod Drop Accident 0.013 delta k

Section	Page
3.3.B.3	3/4 3-3
Bases 3.3.B.3	B3/4 3-3
	B3/4 3-4

The GE Generic Reload Fuel Licensing Topical Report, NEDE-24011-P-A provides justification for removing the 0.013 delta k requirement from the Control Rod Drop Accident evaluation. The justification is given on Page 5-47 of NEDE-24011-P-A and in Section 7-3 of the NRC SER (contained in Appendix C of the topical). The topical indicates it is unrealistic to set a specific value of maximum control rod worth, since this is only one of many parameters in the peak fuel enthalpy calculations for a control rod drop accident. The change does not alter the requirement to satisfy the ultimate 280 cal/gm design limit.

A technical review of these specifications has found them to be acceptable; a safety evaluation has also 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.

SAFETY/RELIEF VALVES

Reactor Coolant System and Automatic Pressure Relief Subsystem

Section	Page
2.2.2.B	2-6
Bases 2.2.1	B2-10
Bases 2.2.2	B2-11
4.5.D.1.a	3/4 5-6
3.5.D.2	3/4 5-7
4.5.D.2	3/4 5-7
3.6.E	3/4 6-5
4.6.E	3/4 6-5
3/4 6.E Bases	B3/4 6-4, B3/4 6-5

These changes to the Technical Specifications envelop and supersede the changes proposed by NNECO in D. C. Switzer's letter to G. Lear dated January 27, 1978. The changes in that previous submittal are, therefore, withdrawn and replaced with the attached changes.

The previous changes submitted by NNECO on January 27, 1978 provided increased surveillance of the Automatic Pressure Relief (APR) and Safety Relief Valves (SRV) in order to improve valve performance and overall plant safety. The attached changes envelop these and also provide, 1) clarification of the surveillance requirements for the APR valves to delete the requirement to mechanically lift the APR's in Section 4.5.D.1.a since manual actuation is required by Section 4.5.D.1.b (consistent with BWR STS), and 2) staggered SRV setpoints to prevent any potential of repeated simultaneous popping, thereby reducing torus loading, and 3) increased SRV setpoints to improve valve simmer margin.

Our technical review of these changes also considered the following:

(1) The effect on the torus of the proposed change of the SRV setpoints has been evaluated.

A. The increased setpoints reduce the probability of spurious actuation of the valves, thus, decreasing the clearing load cycles on the torus. The reduced probability of spurious opening likewise reduces the possibility of a stuck open SRV and any stresses associated with such condition.

B. As reported in GE report NEDC-21581-P, "Monticello Safety Relief Valve Discharge Load Test", the loads on the torus wall for a second actuation (hot pop) of an SRV are greater than for a first actuation of the valve. Furthermore, as reported in the attachments to GE letter MI-G-179, the effect of several valves opening simultaneously is increased loads compared to one valve. The staggered setpoints assure that for various reactor isolation conditions, only one valve would open more than once. The staggered setpoints, therefore, assure that for the hot pop (controlling loading case), only one valve would open; and there would be no additive effect of the hot pop loads.

C. The increased setpoints can result in an increase in the reaction load where the ramshead is attached to the torus ring girder. The magnitude of this load is on the order of 50 kips, and the increase because of a setpoint change would be on the order of 10 percent or 5 kips. This change can then be compared to pool swell down loads and how they have been reduced because of recent testing compared to the Short-Term Program.

The STP reference plant net downward pressure was 21 psi (Figure 3-11, NEDC-20989-P, June, 1976) which resulted in an average torus column load of approximately 650 kips (Teledyne report TR-2138(a), "Plant Unique Analysis", July 26, 1976). More recent testing reported in NEDM-21688-P, "Preliminary Load Evaluation Report", August, 1977, indicates a maximum downward pressure of less than 12 psi. Since the short-term criteria on the torus, in particular, the ring girder and columns, was satisfied for the 21 psi load and since the down load has been significantly reduced to 11 psi, it obviously follows that any possible increase in the ramshead reaction load is less than the reduction in downward pool and, therefore, of no consequence.

(2) The effect on the main steam and relief-valve piping has been evaluated.

A. Teledyne Engineering Services (TES) has reviewed the effects of resetting the actuation points of the Millstone Unit No. 1 Main Steam Safety Relief Valves. The following setpoints were recommended by TES.

Relief Line MS-8a	- no change (1095 psig)
Relief Line MS-8d	- reset to 1110 psig
Relief Line MS-8b, 8c, 8e, and 8f	- reset to 1125 psig

This review was based on a detailed evaluation of dynamic piping stresses, which were done previously by TES.

B. The revised main steam relief valve actuation setpoints have been reviewed to assure that the increased dynamic load does not adversely affect the structural integrity of the main steam and relief valve piping system. All six (6) relief lines and the associated steam lines were analyzed for dynamic loading as a part of the EOC modifications made in 1975. The analytical technique used to calculate the hydrodynamic loading was state-of-the-art at that time, however, more recent techniques indicate the presence of conservatism beyond that anticipated. A margin did exist at that time between the calculated stress and the conservatively based ASME Code limits. A review of this margin and the slightly increased setpoints in light of present-day analytical techniques indicates that the ASME Code limits will not be compromised.

C. It should be noted here, that the addition of quenchers to the relief lines is planned for a future modification. The effects that the quenchers have on the pipe stresses and pressure setpoints will be the topic of a TES analysis scheduled to begin in the immediate future. Additionally, further analysis may have to be initiated to further increase the simmer margin of the SRV's as recommended in GE SIL No. 196, Supplement 3.

The attached NEDO-24168 and NEDO-24168-1, Supplemental Reload Licensing Submittal for Millstone 1, Reload 6, include the staggered and elevated SRV setpoints proposed herein. Thus, the transient and accident response has been evaluated and the results provide assurance that the appropriate limits have not been exceeded with the revised setpoints.

General Electric evaluations of Millstone 1 submitted to NNECO conclude that any grouping of SRV setpoints which have one valve set at least 10 psi below the other five results in only one valve actuating on the second pop. Consequently, the GE parametric study serves to affirm that the proposed setpoints staggered 1-1-4 at 15 psi margin, will prevent a second actuation of all but one valve.

A technical review of these specifications has found them to be acceptable; a safety evaluation has also 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.

DOCKET NO. 50-245

PROPOSED TECHNICAL SPECIFICATION CHANGES

FOR

MILLSTONE 1, RELOAD 6

FEBRUARY, 1979

SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor vessel pressure shall not exceed 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent exceeding the reactor coolant system safety limits.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent exceeding the safety limits.

Specification:

A. Reactor Coolant High Pressure Scram Trip Setting shall be $\leq$ 1085 psig.

B. The safety valve function settings of the six dual purpose relief/safety valves shall correspond with a steam pressure of:

No. of Valves	Set Point (PSIG)
1	1095 $\pm$ 1%
1	1110 $\pm$ 1%
4	1125 $\pm$ 1%

2.1.1 Bases

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than 1.07. MCPR > 1.07 represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation.

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variable, i.e., normal plant operation presented on Figure 2.1.2 by the nominal expected flow control line. The Safety Limit (MCPR of 1.07) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from a normal operating condition more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit (MCPR = 1.07) is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the safety limit are provided at the beginning of each fuel cycle.

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1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958.

Because the boiling transition correlation is based on a large quantity of full scale data there is a very high confidence that operation of a fuel assembly at the condition of MCPR = 1.07 would not produce boiling transition.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Millstone operated above the critical heat flux for a significant period of time (30 minutes) without clad perforation. Thus, although it is not required to establish the safety limit, additional margin exists between the safety limit and the actual occurrence of loss of cladding integrity. The limit of applicability of the boiling transition correlation is 1400 psia during normal power operation. However, the reactor pressure is limited as per Specification 2.2.1.

In addition to the boiling transition limit (MCPR = 1.07) operation is constrained to a maximum LHGR= 17.5 kW/ft for 7 x 7 and 13.4 kW/ft for 8 x 8. At 100% power this limit is reached with a maximum total peaking factor (MTPF) of 3.08 for 7 x 7 fuel and 3.04 for 8 x 8 fuel. For the case of the MTPF exceeding these values operation is permitted only at less than 100% of rated thermal power and only with reduced APRM scram settings as required by Specification 2.1.2.A.1.

At pressures below 800 psia, the core evaluation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 2.1.1A or 2.1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

setting was selected to provide adequate margin from the thermal-hydraulic safety limit and allow operating margin to minimize the frequency of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MTPF and reactor core thermal power. The scram setting is adjusted in accordance with the formula given in Specification 2.1.2A.1, when the MTPF is greater than 3.08 for 7x7 fuel and 3.04 for 8x8 fuel.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR $\geq$ 1.07 when the transient is initiated from MCPR's specified in Section 3.11.C. In order to assure adequate core margin during full load rejections in the event of failure of the select rod insert, it is necessary to reduce the APRM scram trip setting to 90% of rated power following a full load rejection incident. This is necessary because, in the event of failure of the select rod insert to function, the cold feedwater would slowly increase the reactor power level to the scram trip setpoint. A trip setpoint of 90% of rated has been established to provide substantial margin during such an occurrence. The trip setdown is delayed to prevent scram during the initial portion of the transient. The specified maximum setdown delay of 30 seconds is conservative because the cold feedwater transient does not produce significant increases in reactor power before approximately 60 seconds following the load rejection. Reference Amendment 16 Response to Questions A-12, A-14, A-15, and D-3.

For operation in the refuel or startup/hot standby modes while the reactor is at low pressure, the APRM reduced flux trip scram setting of $\leq$ 15% of rated power provides adequate thermal margin between the maximum power and the safety limit, 25% of rated power. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coeffecients are small and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise.

In an assumed uniform rod withdrawal approach to the scram level, the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The APRM reduced trip scram remains active until the mode switch is placed in the run position. This switch occurs when the reactor pressure is greater than 880 psig.

The IRM trip at $\leq$ 120/125 of full scale remains as a backup feature.

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps. During steady-state operation with one recirculation pump operating the equalizer line shall be open. Analyses of transients from this operating condition are less severe than the same transients from the two pump operation.

B. APRM Control Rod Block Trips

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate and thus to protect against a condition of a MCPR < 1.07. This rod block setpoint, which is automatically varied with recirculation flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The specified flow variable setpoint provides substantial margin from fuel damage, assuming steady-state operation at the setpoint, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship. Therefore, the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The total peaking factor assumed for the analysis was 3.08 for 7x7 and 3.04 for 8x8 fuel. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram setting, the APRM rod block setting is adjusted downward according to the equation included in Specification 2.1.2B if peaking factors greater than 3.08 for 7x7 fuel and 3.04 for 8x8 fuel exist; thus preserving the APRM rod block safety margin.

The APRM rod block setpoint is reduced to $\leq$ 12% of rated thermal power with the mode switch in refuel or Startup/Hot Standby position.

C. Reactor Low Water Level Scram

The reactor low water level scram is set at a point which will assure that the water level used in the basis for the safety limit is maintained.

D. Reactor Low Low Water Level ECCS Initiation Trip Point

The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the decay heat associated with the loss-of-coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish this function, the capacity of each emergency core cooling system component was established based on the reactor low low water level. To lower the setpoint of the low water level scram would require an increase in the capacity of each of the ECCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet will not be set lower because of ECCS capacity requirements.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS would not prevent the ECCS components from meeting their design criteria. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Scram

The turbine stop valve scram like the load rejection scram anticipates the pressure, neutron flux and heat flux increase caused by the rapid closure of the turbine stop valves and failure of the bypass. With a scram setting $\leq$ 10% of valve closure the resultant increase in surface heat flux is limited such that MCPR remains above 1.07 even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is < 45% of rated, as measured by the turbine first stage pressure.

F. Turbine Control Valve Fast Closure

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass; i.e., it prevents MCPR from becoming less than 1.06 for this transient. For the load rejection from 100% power, the heat flux increases to only 106.5% of its rated power value which results in only a small decrease in MCPR. This trip is bypassed below a generator output of 307 MWe because, below this power level, the MCPR is greater than 1.07 throughout the transient without the scram.

In order to accommodate the full load rejection capability, this scram trip must be bypassed because it would be actuated and would scram the reactor during load rejections. This trip is automatically bypassed for a maximum of 260 millisec following initiation of load rejection. After 260 millisec, the trip is bypassed providing the bypass valves have opened. If the bypass valves have not opened after 260 millisec, the bypass is removed and the trip is returned to the active condition. This bypass does not adversely affect plant safety because the primary system pressure is within limits during the worst transient even if this trip fails. There are many other trip functions which protect the system during such transients. Reference Response D-3 of Amendment 16.

2.2.1 Bases:

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, coolant system piping and isolation condenser. The respective design pressures are 1250 psig at 575°F, 1175 psig at 564°F, and 1250 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section IIIfor the pressure vessel and isolation condenser and USAS B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the primary coolant system.

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is more than a factor of 1.5 below the yield strength of 43,300 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

The relationships of stress levels to yield strength are comparable for the isolation condenser and primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1035 psig. For the turbine trip or loss of electrical load transients the turbine trip scram or generator load rejection scram, together with the turbine bypass system limits the pressure to less than 1085 psig. The safety/relief valves are set at 1095 psig, 1110 psig, and 1125 psig and are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for the turbine bypass system. Credit is taken for the neutron flux scram, however.

During operation, reactor pressure is continuously displayed in the control room on a 0-1500 psig pressure recorder.

2.2.2 Bases

In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the specified settings of the pressure relieving devices are below 103% of design pressure. As described in the General Electric Topical Report, NEDE-24011-P-A, Generic Reload Fuel Application (Section 5.3), the most severe isolation event with indirect scram has been evaluated. The most severe isolation is the MSIV closure from steady-state operation at 2011 MW_t. The evaluation assures that the sizing and settings of the pressure relieving devices is adequate to assure that the peak allowable pressure of 110% of vessel design pressure is not exceeded.

Evaluations indicate that a total of six dual purpose safety/relief valves set at the specified pressures maintain the peak pressure during the transient well within the code allowable and safety limit pressure.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.B Control Rod Withdrawal 3. Whenever the reactor is in the startup or run mode below 20% rated thermal power, no control rods shall be moved unless the rod worth minimizer is operable or a second independent operator or engineer verifies that the operator at the reactor console is following the control rod program. The second operator may be used as a substitute for an inoperable rod worth minimizer during a startup only if the rod worth minimizer fails after withdrawal of at least twelve control rods. 4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.	4.3.B Control Rod Withdrawal 3. (a) To consider the rod worth minimizer operable, the following steps must be performed: (i) The control rod withdrawal sequence for the rod worth minimizer computer shall be verified as correct. (ii) The rod worth minimizer compute line diagnostic test shall be successfully completed. (iii) Proper annunciation of the select error of at least one out-of-sequence control rod in each fully inserted group shall be verified. (iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point. (b) If the rod worth minimizer is inoperable while the reactor is in the startup or run mode below 10% rated thermal power, and a second independent operator or engineer is being used, he shall verify that all rod positions are correct prior to commencing withdrawal of each rod group.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3. From and after the date that the FWCI subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Pressure Relief Subsystem, the core spray subsystems, LPCI subsystem, and isolation condenser system are operable.

4. If the requirements of 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

D. Automatic Pressure Relief (APR) Sybsystems

1. Except as specified in 3.5.D.2 and 3 below, the APR subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel.

3. When it is determined that FWCI subsystem is inoperable, the LPCI subsystem, both core spray subsystems, the automatic pressure relief subsystems and the motor operated isolation valves and shell side makeup system for the isolation condenser system shall be demonstrated to be operable immediately. The automatic pressure relief subsystem and motor operated isolation valves and shell side makeup system of the isolation condenser shall be demonstrated to be operable daily thereafter.

D. Surveillance of the Automic Pressure Relief Subsystem shall be performed as follows:

1. During each operating cycle, the following shall be performed:

    a. A simulated automatic initiation of the system throughout its operating sequence but excludes actual valve opening, and

    b. With the reactor at low pressure, each relief valve shall be manually opened until valve operability has been verified by torus water level instrumentation, or by an audible discharge detected by an individual located outside the torus in the vicinity of each relief line.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

2. From and after the date that one of the three relief/safety valves of the automatic pressure relief subsystem is made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding seven days unless repairs are made and provided that during such time the remaining automatic pressure relief valves, FWCI subsystem and gas turbine generator are operable.

3. If the requirements of 3.5.D cannot be met, an orderly reactor shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

E. Isolation Condenser System

1. Whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel, the isolation condenser shall be operable except as specified in 3.5.E.2 and the shell side water level shall be greater than 66 inches.

2. When it is determined that one safety/relief valve of the automatic pressure relief subsystem is inoperable the actuation logic of the remaining APR valves and FWCI subsystem shall be demonstrated to be operable immediately and daily thereafter.

E. Surveillance of the Isolation Condenser System shall be performed as follows:

1. Isolation Condender System Testing:

   a. The shell side water level and temperature shall be checked daily.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

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 and whenever the reactor coolant pressure is greater than 90 psig, and temperature greater than 320°F, the safety valve function of the six relief/safety valves shall be operable. (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 and temperature below 320°F within 24 hours.

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

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 $\pm$ 1%
1	1110 $\pm$ 1%
4	1125 $\pm$ 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 an audible discharge detected by an individual located outside the torus in the vicinity of each discharge.

F. Structural Integrity

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

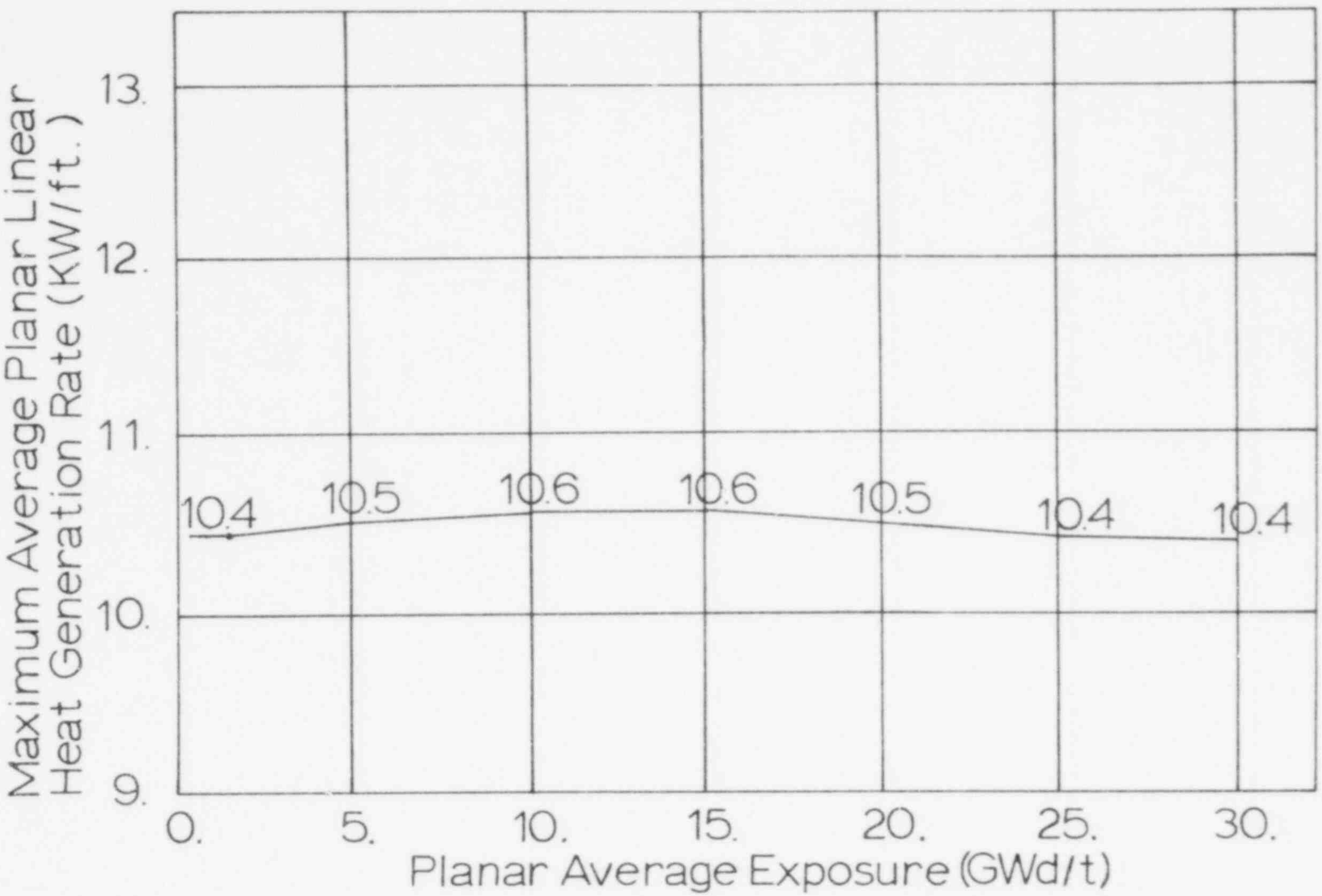

Figure 3.11.1h MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE. RELOAD 6 (8DRB265H and 8DRB265L)

Amendment no.

TABLE 3.11.1

OPERATING LIMIT MCPR'S FOR CYCLE 7

Core Average Burn-up Range	Operating Limit MCPR 7 x 7 Fuel	Operating Limit MCPR 8 x 8 Fuel
BOC7 to EOC7	1.27	1.34
Coastdown beyond EOC7 (100% power to 70% power) (Restricted to 100% flow)	1.27	1.34

End of Cycle is defined as end of full power life for the cycle

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 trip settings of 127 inches of water and 79 inches of water 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 on update of the analysis. Below $\sim$ 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.

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, "GE/BWR Generic Reload Application for 8 x 8 Fuel, Rev. 1, Supplement 4 (NEDO-20360)," 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 limiting parameters.

a. A startup inter-assembly local power peaking factor of 1.30 or less.(5)
b. An end of cycle delayed neutron fraction of 0.005.
c. A beginning of life Doppler reactivity feedback.
d. The Technical Specification rod scram insertion rate.
e. The maximum possible rod drop velocity (3.11 ft/sec).
f. The design accident and scram reactivity shape function.
g. The moderator temperature at which criticality occurs.

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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.

(5) To include the power spike effect caused by gaps between fuel pellets.

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-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Reference Section 7-9 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.

For a crack size which gives a leakage rate of 2.5 gpm, the probability of rapid propagation is less than 10^{-5}. A leakage rate of 2.5 gpm is detectable and measurable.

The 25 gpm limit on total leakage to the containment was established by considering the removal capabilities of the pumps. The capacity of either of the drywell sump pumps is 50 gpm and the capacity of either of the drywell equipment drain tank pumps is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

The performance of the reactor coolant leak detection system will be evaluated during the first year of commercial operation and the conclusions of this evaluation will be reported to the AEC.

The main steam line tunnel leakage detection system is capable of detecting small leaks. The system performance will be evaluated during the first five years of plant operation and the conclusions of the evaluation will be reported to the AEC.

E. Safety and Relief Valves

Present experience with the new safety/relief valves indicates that a testing of at least 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as $\pm 1\%$ of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded.

The relief/safety valves have two functions; i.e., power relief or self-actuated by high pressure. The solenoid actuated function (automatic pressure relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate the valves to opcn. This function is discussed in Specification 3.5.D. In addition, the valves can be operated manually.

The safety function is performed by the same relief/safety valve with a pilot valve causing main valve operation.

When the setpoint is being bench checked, it is prudent to disassemble one of the relief/safety valves to examine for crud buildup, bending of certain actuator members,or other signs of possible deterioration.

Testing at low reactor pressure is required during each operating cycle. It has been demonstrated that the blowdown of the valve to the torus causes a wave action that is detectable on the torus water level instrumentation. The discharge of a safety valve is audible to an individual located outside the torus in the vicinity of the line, as experienced at other BWR's.

F. Structural Integrity

A preservice inspection of the components listed in Table 4.6.1 will be conducted to establish a reference base for later inspections. Construction oriented nondestructive testing is being conducted as systems are fabricated to assure freedom from defects greater than code allowance. In addition, the facility has been designed such that defects greater than code should not occur throughout plant life. Information concerning the structural integrity of the reactor pressure vessel can be found in Appendix E to the FSAR. This Appendix contains documentation of design, fabrication, inspection, analysis and testing of this pressure vessel.

Design confirmation and construction adequacy will be demonstrated during the plant startup and power ascension test program. As part of this program, cold and hot vibration tests on certain reactor vessel internals will be performed. The tests, described in Amendments 17 and 18, are designed to obtain data on the unique design features of Millstone Unit 1 as compared to Dresden Unit 2 design. Thus, the basis for the Millstone vibration test program is predicted on obtaining necessary data to confirm common design features shared with earlier BWR plants such as Dresden Unit 2. In the event that data from these earlier plants are not available before routine power operation of Millstone Unit 1, the matter will be reviewed as indicated in Amendment No. 23.

In order to monitor the integrity of the primary pressure boundaries throughout plant life, the inspection program stated in Table 4.6.1 was developed. This program was developed using the ASME Code for In-Service Inspection as a basis and provides for exclusion of certain inspection parameters where current technology does not provide a means of inspecting or where equipment access and radiation hazards are of greatest significance. The initial inspection program was developed by Northeast Utilities Service Company with assistance from Southwest Research Institute and Teledyne Materials Research. In early 1966, initial efforts were made to establish a nondestructive testing program for reactor vessel surveillance. Shortly after this program initiation, the services of Southwest Research Institute and Lessels Associates, now Teledyne Materials Research, were retained to aid in this program development. After considerable effort of all parties, the initial inspection program for Millstone Unit 1 was finalized in November 1968; this was one month after issuance of the first draft of the Code for In-Service Inspection of Nuclear Reactor Coolant Systems.