DOCKET NO. 50-245

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION CHANGES

FOR

MILLSTONE 1, RELOAD 7

SEPTEMBER, 1980

8009300 311

1. MAPLHGR Limits

## Description of Changes and Safety Evaluation Summary

The Maximum Average Planar Linear Heat Generation Rate limits proposed herein, are based on accident analysis performed with the Cycle 8 core. The limits are supported by the enclosed topical report, NEDO-24085-1 (Attachment 2).

The existence of a gas turbine emergency power supply instead of a second diesel makes Millstone Unit No. 1 unique with regard to single-failure considerations in the LOCA analysis. Accordingly, the entire recirculation line break spectrum (both large and small breaks) has been analyzed in NEDO-24085-1, and reference to a lead plant is not necessary.

The small break analysis is consistent with the current Technical Specification bases contained in Amendment No. 67 to License No. DPR-21, transmitted by letter dated May 8, 1980, D. M. Crutchfield to W. G. Counsil. The addition of a fourth ADS valve to improve depressurization and reflooding capability is the only significant change.

A technical review of these changes 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.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

## 3.11 REACTOR FUEL ASSEMBLY

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

1. During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figure 3.11.1.

2. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR specified in Section 3.11.A.1 is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

## 4.11 REACTOR FUEL ASSEMBLY

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operation conditions.

Objective

The Objective of Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

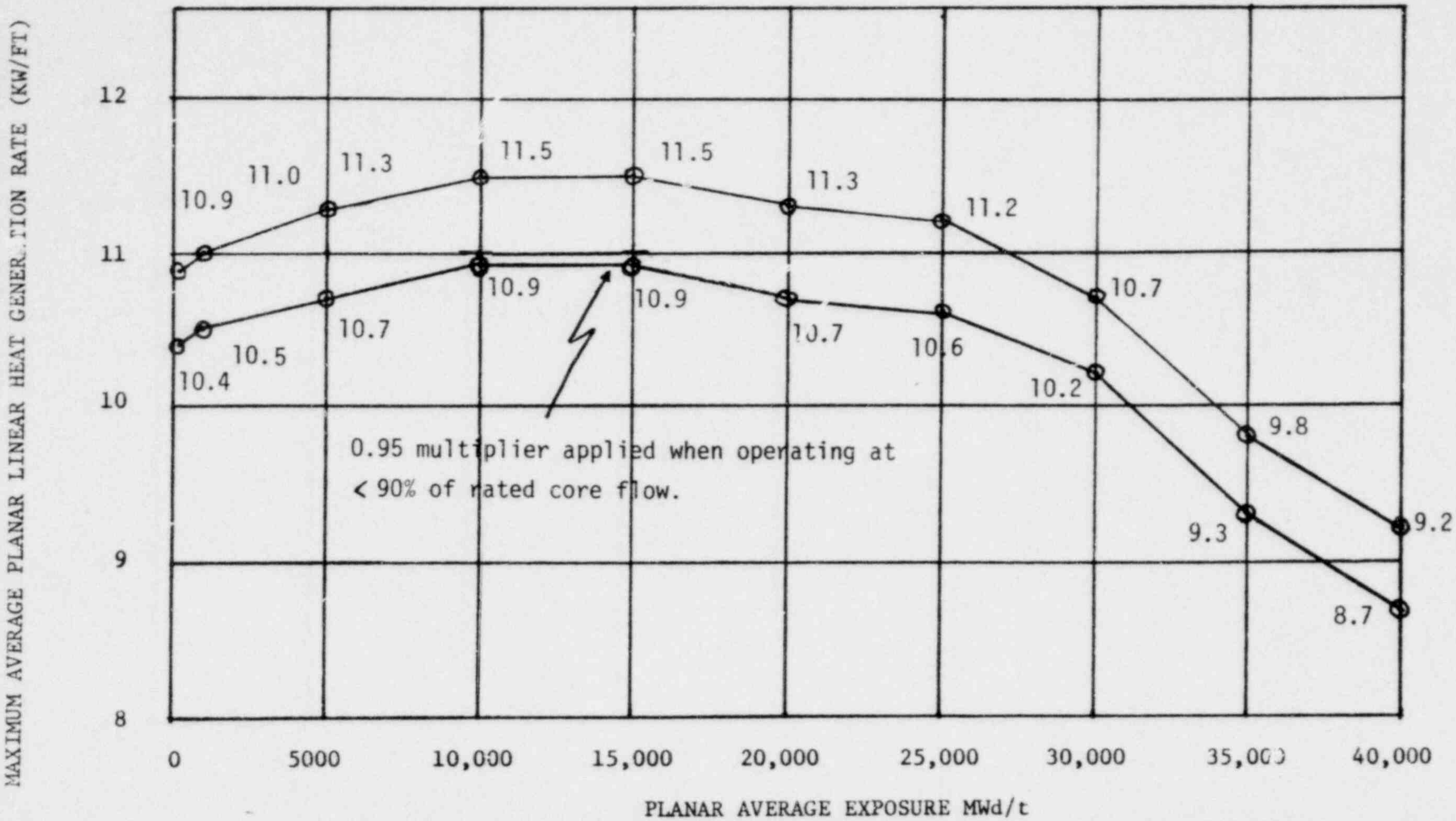


Figure 3.11.1a - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE 8DB262.

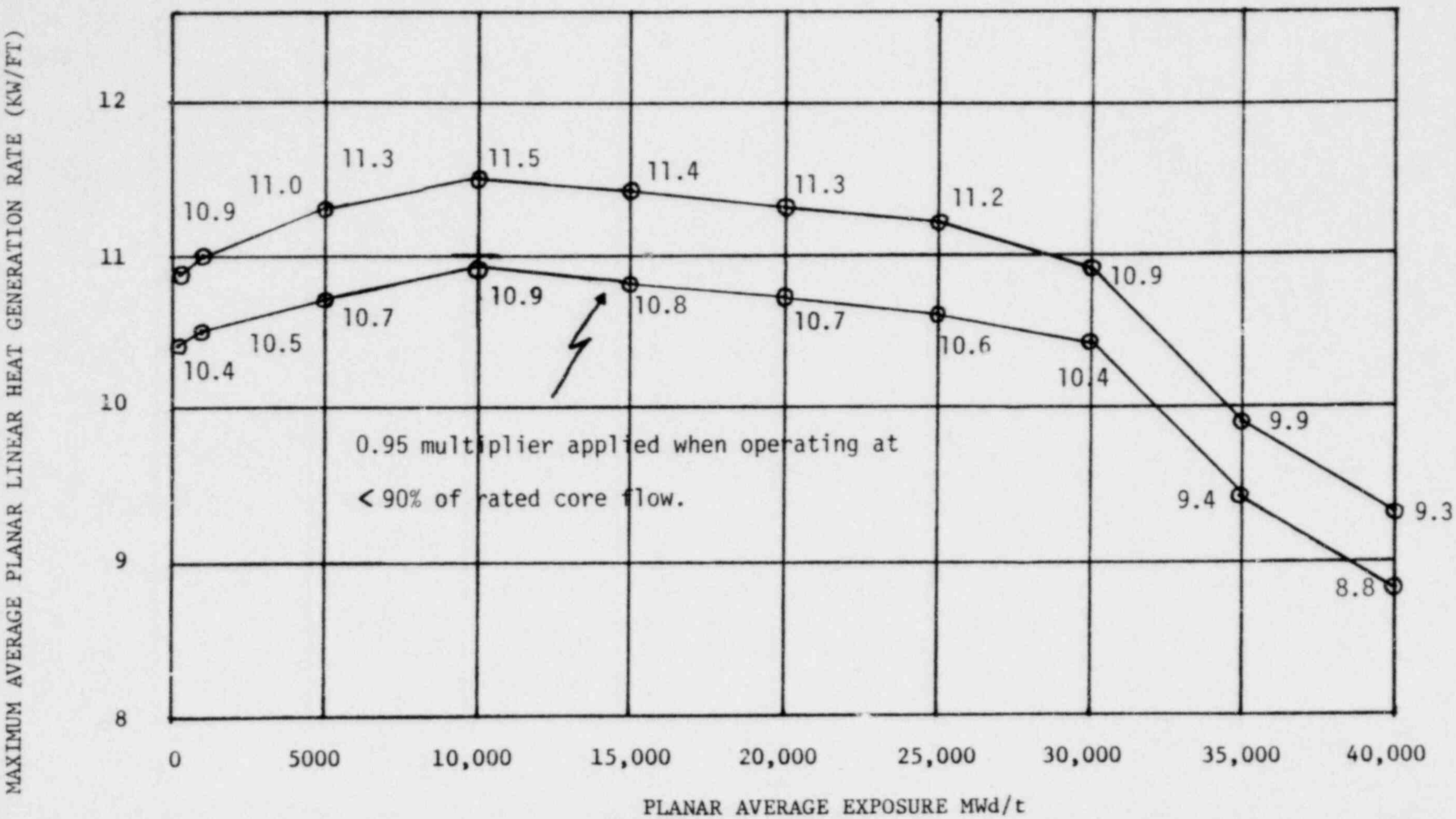


Figure 3.11.1b - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE 8D274L.

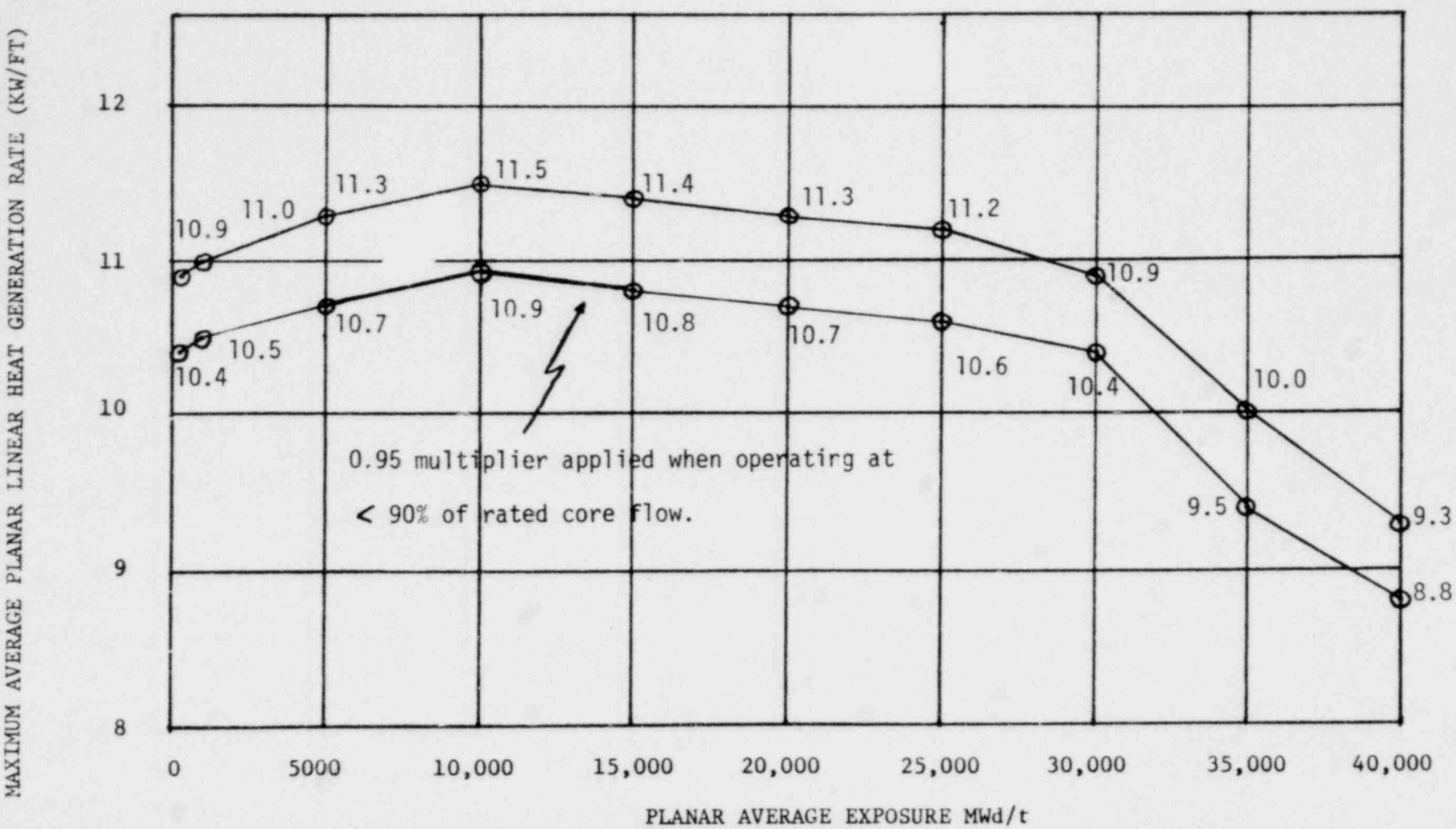


Figure 3.11.1c - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE 8D274H.

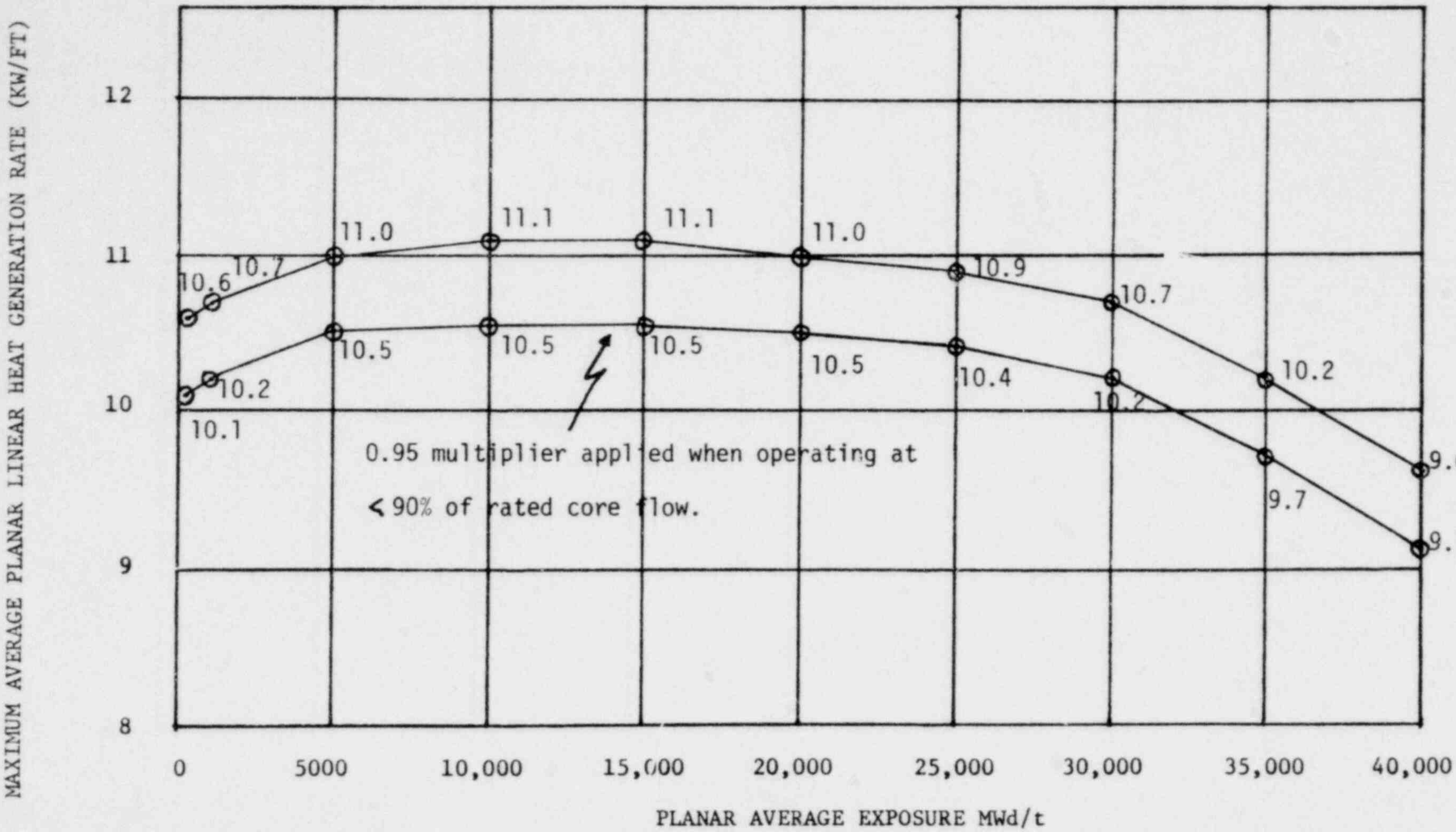


Figure 3.11.1d - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE 8DR265L.

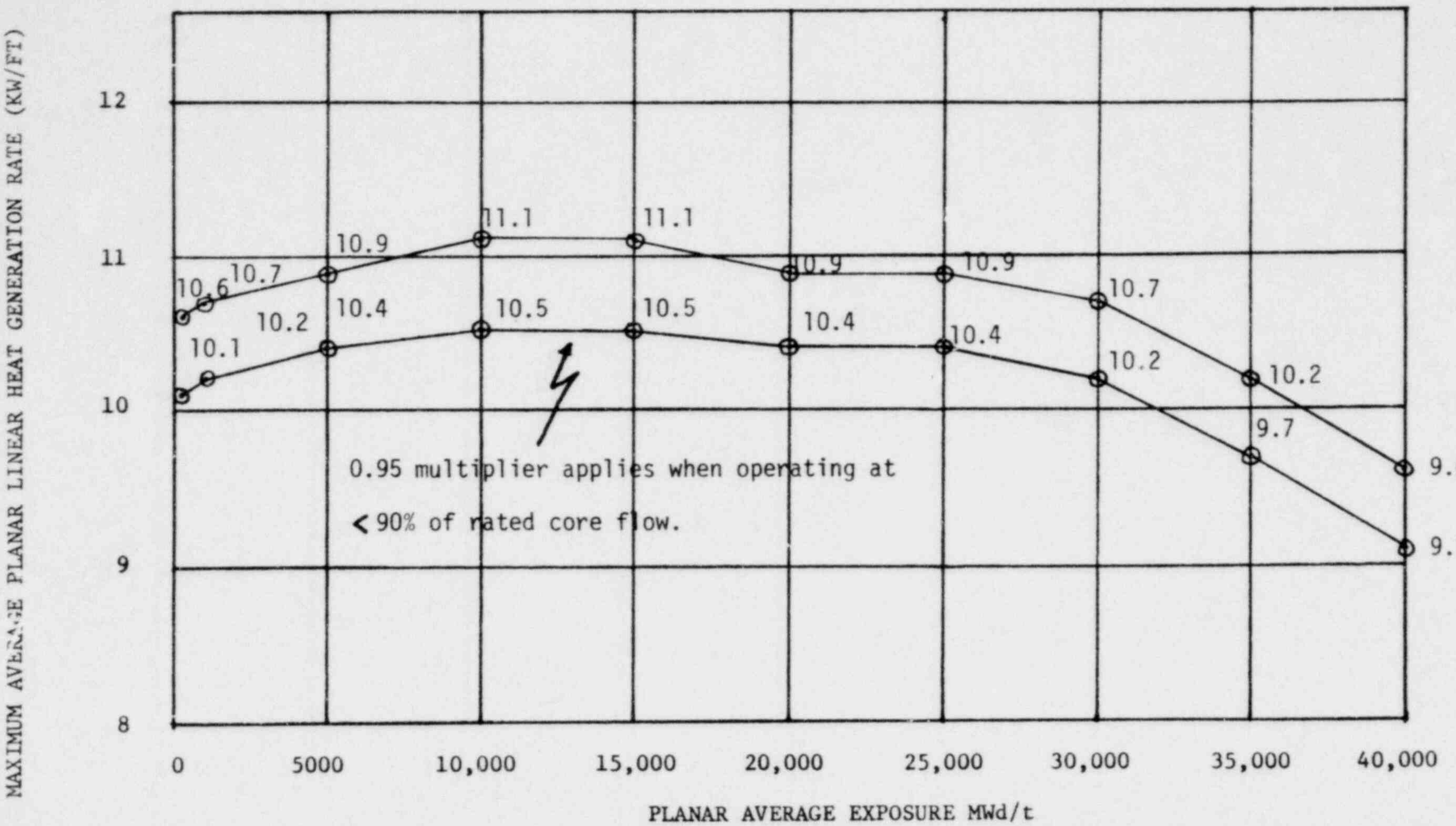


Figure 3.11.1e - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE 8DR265H.

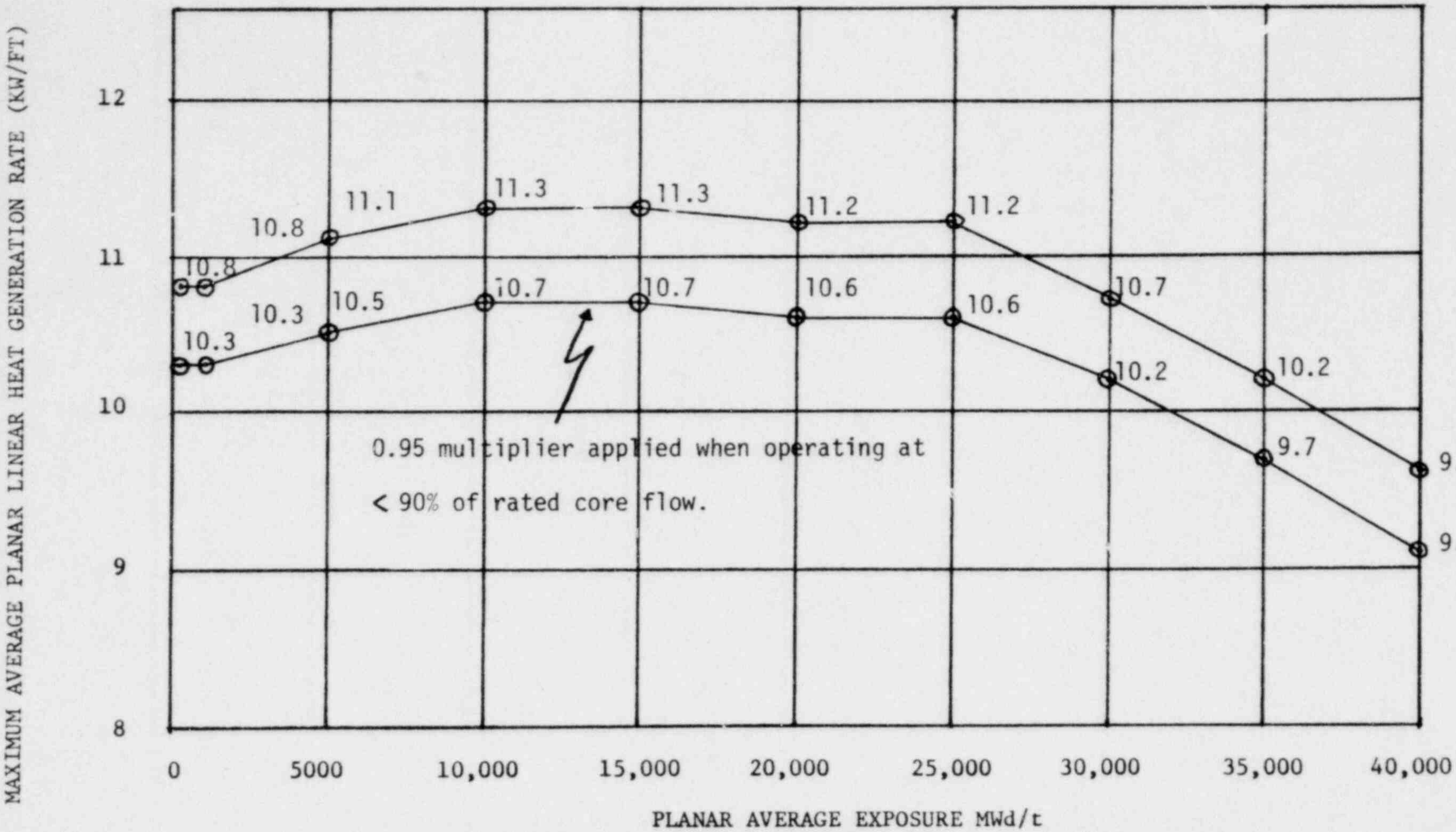


Figure 3.11.1f - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE P8DR265H.

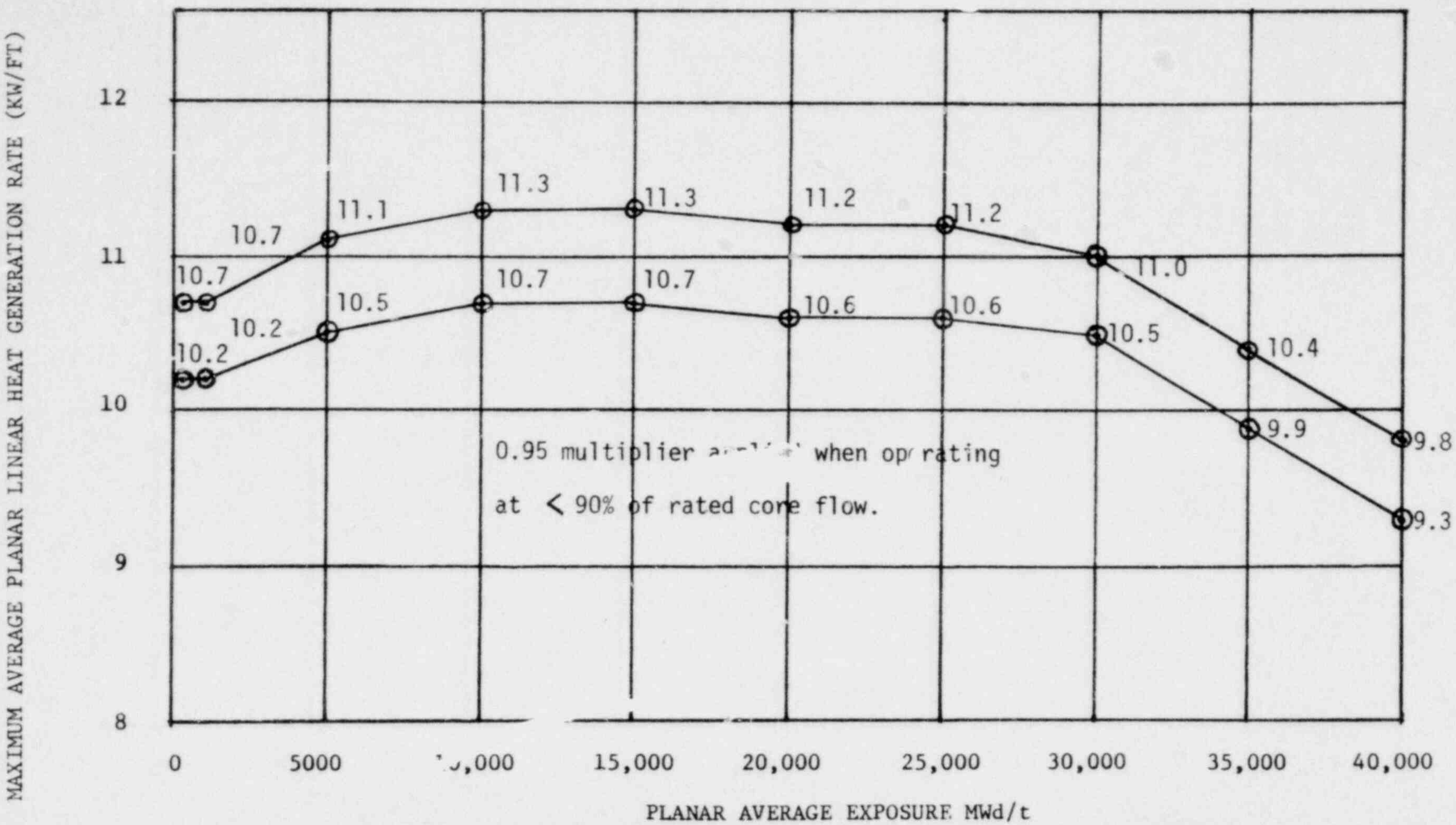


Figure 3.11.1g - MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS PLANAR AVERAGE EXPOSURE. FUEL TYPE P8DR282.

3.11 and 4.11 Bases

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10 CFR 50 Appendix K limit. The limiting value for APLHGR is shown in Figure 3.11.1.

Conservative LOCA calculations predict that nucleate boiling will be maintained for several seconds following a design basis LOCA. This results in early removal of significant amounts of stored energy which, if present later in the transient, when heat transfer coefficients are considerably lower, would result in higher peak cladding temperature. As core flow is reduced below about 90%, the time of onset of boiling transition makes a sudden change from greater than about 5 seconds to less than 1 second. The approved ECCS evaluation model requires that at the first onset of local boiling transition, the severely reduced heat transfer coefficients must be applied to the affected planar area of the bundle, and thus exaggerates the calculated peak clad temperature. The effect is to significantly reduce the energy calculated to be removed from the fuel during blowdown. This results in an increase in calculated peak clad temperature of about 100°F which can be offset by a 5% reduction in MAPLHGR. For flows less than 90% of rated, a 5% reduction in the MAPLHGR limits in Figure 3.11.1, derived for 100% flow will assure that the plant is operated in compliance to 10 CFR 50.46 at those lower flows.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of References 1 and in Reference 2, 3, and 4 and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% power to determine if fuel

2. New Multiplier for APRM Rod Block Monitor Setting

Description of Changes and Safety Evaluation Summary

The current specification uses the factor [A/MTPF], when the total peaking factor is above the design value, as a multiplier on the peaking factor modification term of the APRM scram and rod block equations. It is proposed that this term be replaced with an equivalent term [FRP/MFLPD]. "FRP" is the fraction of rated power, and "MFLPD" is the Maximum Fraction of Limiting Power Density. This new term will eliminate the need to revise the $P_1$ Program for changes in bundle design, and also insures consistency in process computer calculations.

A technical review of these changes 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.

K. Operating

Operating means that a system or component is performing its intended function in its required manner.

L. Operating Cycle

Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

M. Fraction of Limiting Power Density

The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type. Design LHGR's are 13.4 KW/ft for 8x8, 8x8R and P8x8R bundles.

Maximum Fraction of Limiting Power Density

The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

N. Primary Containment Integrity

Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied.

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in the airlock is closed and sealed.

3. All automatic containment isolation valves are operable or are deactivated in the isolation position.

4. All blind flanges and manways are closed.

O. Protective Instrumentation Definitions

1. Instrument Channel

An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by the instrument channel.

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.2A and a control rod scram does not occur.

D. Whenver the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core. This level shall be continuously monitored.

where:

S = Setting in percent of rated thermal power (2011 MWt)

W = Total recirculation flow in percent of design ($29.7 \times 10^{-6}$ lbm/hr)

The trip setting shall not exceed 90 percent of rated power during generator load rejections from an initial generator power greater than 307 MWe. The APRM scram setdown shall be 90% of rated within 30 seconds after initiation of full load rejection.

b. In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.65\ W + 55\% \left[\frac{FRP}{MFLPD}\right]$$

where,

FRP = fraction of rated thermal power (2011 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8x8, 8x8R and P8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

c. During power ascensions with power levels less than or equal to 90%, APRM Flux Scram Trip Setting adjustments may be made as described below, provided that the change in scram setting adjustment is less than 10% and, a notice of the adjustment is posted on the reactor control panel.

The APRM meter indication is adjusted by:

$$APRM = \left(\frac{MFLPD}{FRP}\right) P$$

where:

APRM = APRM Meter Indication

P = % Core Thermal Power

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

2. <u>APRM Reduced Flux Trip Setting (Refuel or Startup/Hot Standby Mode)</u>

When the mode switch is in the refuel or Start Up/Hot Standby position, the APRM scram shall be setdown to less than or equal to 15% of rated thermal power. The IRM scram trip setting shall not exceed 120/125 of full scale.

B. 1. <u>APRM Rod Block Trip Setting</u>

a. The APRM rod block trip setting shall be as shown in Figure 2.1.2 and shall be: (Run Mode)

$S_{RB} \leq 0.65\ W + 42\%$

where:

$S_{RB}$ = Rod block setting in percent of rated thermal power (2011 MWt).

W = Total recirculation flow in percent of design ($29.7 \times 10^6$ lbm/hr).

b. In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S_{RB} \leq (0.65\ W + 42\%) \left[\frac{FRP}{MFLPD}\right]$$

where:

FRP = fraction of rated thermal power (2011 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 KW/ft for 8x8, 8x8R and P8x8R fuel.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

c. During power ascensions with power levels less than or equal to 90%, APRM Rod Block Trip Setting adjustments may be made as described below, provided that the change in scram setting adjustment is less than 10% and a notice of the adjustment is posted on the reactor control panel:

The APRM meter indication is adjusted by:

$$ARPM = \left(\frac{MFLPD}{FRP}\right)\ P$$

where:

APRM = APRM Meter Indication

P = % Core Thermal Power

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.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.2.A.1, when the maximum fraction of limiting power density is greater than the fraction of rated power. If the APRM scram setting should require a change due to an abnormal peaking condition, it will be done by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced scram curve by the reciprocal of the APRM gain change.

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.

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 coefficients 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 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 actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system.

When the maximum fraction of limiting power density exceeds the fraction of rated thermal reactor power, the rod block setting is adjusted in accordance with the formula in Specification 2.1.2.B. If the APRM rod block setting should require a change due to an abnormal peaking condition, it will be done by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced rod block curve by the reciprocal of the APRM gain change.

The APRM rod block setpoint is reduced to ≤ 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 bases 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.

3. Acoustic Monitoring for Safety/Relief Valves

Description of Changes and Safety Evaluation Summary

Acoustic monitors were installed on the discharge lines of all six safety/relief valves at Millstone Unit No. 1, in December, 1979, consistent with the NRC position in NUREG-0578. This post-TMI plant improvement provides additional capability to verify safety/relief valve operation, in addition to the methods already stipulated in the Technical Specifications. The acoustic monitoring system provides no control action, but only verification information.

A technical review of these changes 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.

| 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.

3. When the safety/relief valves are required to be operable per Specification 3.6.E.1, the Acoustic Valve Postion 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.

4. If Specification 3.6.E.3 is not met, reactor operation is permissible only for the succeeding 30 days unless the Acoustic Valve Position Indication System is made operable sooner.

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 Acoustic Valve Position Indication System, or by an audible discharge detected by an individual located outside the torus in the vicinity of each discharge.

4. The Acoustic Valve Position Indication System shall be functionally tested once every three months, and calibrated once per operating cycle.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| F. Structural Integrity<br><br>The structural integrity of the primary boundary shall be maintained as specified in Technical Specification 3.13. | F. Structural Integrity<br><br>Inservice Inspection and Testing of primary system boundary components shall be performed as specified in Surveillance Requirement 4.13. |

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 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 open. 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.

It is understood that portions of the Acoustic Valve Position Indication cannot be repaired or replaced during operation, therefore, the plant must be shutdown to accomplish such repairs. The 30-day period to do this allows the operator the flexibility to choose his time for shutdown; meanwhile, because of the redundancy provided by the valve discharge temperature monitor and the continued monitoring of the remaining valves by both methods, the ability to detect the opening of a safety/relief valve would not be compromised. The valve operability is not affected by failure of the Acoustic Valve Position Indication System.

4. MCPR Limits

Description of Changes and Safety Evaluation Summary

The Minimum Critical Power Ratios proposed herein, are based on transient analysis performed with the Cycle 8 core. The limits are supported by the enclosed topical report, Y1003J01A09, (Attachment 1).

Since the nominal Cycle 8 exposure is 7,800 MWd/t, with a significant change in MCPR during the Cycle, exposure dependent MCPR's are proposed in Technical Specification Table 3.11.1, thus, allowing more operational flexibility early in the Cycle. The 6,000 MWd/t in Table 3.11.1 corresponds to EOC8-1800 MWd/t on Page 3 of the topical report, Y1003J01A09.

The applicability of end-of-cycle MCPR values during coastdown to 70% power is presented in NEDE-24011-P-A, Section 5.2.

A technical review of these changes 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.

## TABLE 3.11.1

### OPERATING LIMIT MCPR'S FOR CYCLE 8

| BOC8 TO 6000 MWd/t | 6000 MWd/t TO EOC8 | EOC8 TO 70% COASTDOWN | FUEL TYPE |
|---|---|---|---|
| 1.31 | 1.37 | 1.37 | 8 x 8 |
| 1.29 | 1.37 | 1.37 | 8 x 8R |
| 1.31 | 1.39 | 1.39 | P8 x 8R |

5. Fourth ADS Valve

Description of Changes and Safety Evaluation Summary

An additional one, of the six existing safety/relief valves, will be added to the automatic depressurization system (ADS), also known as automatic pressure relief system (APR), by modification of the actuation logic. The result is that four of six S/RV's will open for the ADS function instead of three of six.

The reason for the change is to improve ECCS response to a small break LOCA by causing a more rapid vessel depressurization if ADS is required. This is only needed in the event of a loss of feedwater combined with a loss of FWCI.

The following impacts were considered in the modification to add a fourth valve to ADS:

-- torus loads
-- preferred valve/discharge piping location
-- torus heat-up
-- S/RV discharge piping loads
-- reactor vessel cooldown
-- impact of depressurization on reactor internals
-- revised ECCS performance for small break LOCA

In all cases, the impact of the change was within design criteria of the affected equipment. Therefore, the impact of the change is negligible. The ECCS response improvement was fairly significant such that SBA is maintained less limiting than DBA, even though partial LPCI flow is lost through the assumed loop selection logic error.

The overall change is beneficial to plant safety with no significant adverse effects.

The addition of a fourth ADS valve eliminates a planar heat generation limit that was imposed due to concerns over certain small break scenarios. The concern and the restrictions were previously reported and it was noted at the time that additional ADS credit was a method that could be utilized to alleviate the small break concerns.

A technical review of these changes 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.

## LIMITING CONDITION FOR OPERATION

2. From and after the date that one of the four 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. <u>Isolation Condenser System</u>

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.



## SURVEILLANCE REQUIREMENT

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.

without the use of off-site electrical power. For the pipe breaks for which the FWCI is intended to function, the core never uncovers and is continuously cooled and thus no clad damage occurs. The repair times for the limiting conditions of operation were set considering the use of the FWCI as part of the emergency core cooling system and isolation cooling system.

The FWCI utilizes portions of the normally operating feedwater system; e.g., condensate, condensate booster and feedwater pumps. Therefore, the reliability of the pumps, valves and motors is constantly being demonstrated. Thus the system has an inherently higher degree of reliability than normally non-operating systems. Since an operating string of pump and valves is programmed for FWCI operation, it is not expected that the normally operating portions of the FWCI would be out of operation during normal operation. Thus, to demonstrate the operability of the FWCI, it is usually sufficient to demonstrate the non-operating portions of the system.

D. Automatic Pressure Relief (APR)

The relief valves of the automatic pressure relief subsystem are a back-up to the FWCI subsystem. They enable the core spray or LPCI to provide protection against the small pipe break in the event of FWCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays or LPCI. The core spray and/or LPCI provide sufficient flow of coolant to limit fuel clad temperatures to well below clad melt and to assure that core geometry remains intact.

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

E. Isolation Condenser System

The turbine main condenser is normally available. The isolation condenser is provided for core decay heat removal following reactor isolation and scram. The isolation condenser has a heat removal capacity sufficient to handle the decay heat production at 300 seconds following a scram. Water will be lost from the reactor vessel through the relief valves in the 300 seconds following isolation and scram. This represents a minor loss relative to the vessel inventory.

6. Jet Pump Baseline Data

Description of Changes and Safety Evaluation Summary

This proposed change deletes the requirement to obtain single-loop flow Jet Pump baseline data, since single-loop operation is not licensed nor permitted at Millstone Unit No. 1. The existing Surveillance Requirement is, thus, pointless.

A technical review of this change has found it to be acceptable; a safety evaluation has also been performed in accordance with 10CFR50.59 and has concluded that this change does not constitute an unreviewed safety question. The Millstone Nuclear Review Board has reviewed and approved the proposed change and has concurred with the above determination.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

G. Jet Pumps

1. Whenever the reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication cannot be obtained for at least nineteen jet pumps, an orderly shutdown shall be initiated within 12 hours and the reactor shall be in a cold shutdown condition within 24 hours.

H. Recirculation Pump Flow Mismatch

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

G. Jet Pumps

1. Whenever there is a recirculation flow with the reactor in the startup/hot standby or run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.

3. The baseline data required to evaluate the conditions in Specifications 4.6.G.1 will be acquired each operating cycle.

H. Recirculation pump speed shall be checked daily for mismatch.

7. Primary Containment Isolation Valves

Description of Changes and Safety Evaluation Summary

These proposed chang.s are administrative in nature, and only correct several typographical errors. No actual valve position changes are resulting from these proposed corrections, and they only reflect the plant design that has been in effect, proven workable, and safe for the plant operating history.

A technical review of these changes 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.

## TABLE 3.7.1

## PRIMARY CONTAINMENT ISOLATION

| Isolation Group | Valve Identification (Valve Number) | Number of Power Operated Valves Inboard | Number of Power Operated Valves Outboard | Maximum Operating Time (Sec) | Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| 1 | Main Steam Line Isolation (MS-1A, 2A, 1B, 2B, 1C, 2C, 1D, 2D) | 4 | 4 | $3 \leq T \leq 5$ | O | GC |
| 1 | Main Steam Line Drain (MS-5) | 1 | | 35 | C | SC |
| 1 | Main Steam Line Drain (MS-6) | | 1 | 35 | C | SC |
| 1 | Recirculation Loop Sample Line (SM-1, 2) | 1 | 1 | 5 | C | SC |
| 1 | Isolation Condenser Vent to Main Steam Line IC-6, 7) | | 2 | 5 | O | GC |
| 2 | Drywell Floor Drain (SS-3, 4) | | 2 | 20 | O | GC |
| 2 | Drywell Equipment Drain (SS-13, 14) | | 2 | 20 | O | GC |
| 2 | Drywell Vent (AC-7) | | 1 | 10 | C | SC |
| 2 | Drywell Vent Relief (AC-9) | | 1 | 15 | C | SC |
| 2 | Drywell and Suppression Chamber Vent from Reactor Building (AC-8) | | 1 | 10 | C | SC |
| 2 | Drywell Vent to Standby Gas Treatment System (AC-10) | | 1 | 10 | C | SC |
| 2 | Suppression Chamber Vent (AC-11) | | 1 | 10 | C | SC |
| 2 | Suppression Chamber Vent Relief (AC-12) | | 1 | 15 | C | SC |
| 2 | Suppression Chamber Supply (AC-6) | | 1 | 10 | C | SC |
| 2 | Drywell Supply (AC-5) | | 1 | 10 | C | SC |
| 2 | Drywell and Suppression Chamber Supply (AC-4) | | 1 | 10 | C | SC |
| 3 | Cleanup Demineralizer System (CU-2) | 1 | | 18 | O | GC |
| 3 | Cleanup Demineralizer System (CU-3, 28) | | 2 | 18 | O | GC |
| 3 | Shutdown Cooling System (SD-1) | 1 | | 48 | C | SC |
| 3 | Shutdown Cooling System (SD-2A, 2B, 4A, 4B) | | 4 | 48 | C | SC |
| 3 | Shutdown Cooling System (SD-5) | | 1 | 48 | C | SC |
| 3 | Reactor Head Cooling Line (HS-4) | | 1 | 45 | C | SC |
| 4 | Isolation Condenser Steam Supply (IC-1) | 1 | | 24 | O | GC |
| 4 | Isolation Condenser Steam Supply (IC-2) | | 1 | 24 | O | GC |
| 4 | Isolation Condenser Condensate Return (IC-3) | 1 | | 19 | C | SC |
| 4 | Isolation Condenser Condensate Return (IC-4) | | 1 | 19 | O | GC |
| | Feedwater Check Valves (FW-9A, 10A, 9B, 10B) | 2 | 2 | NA | O | Process |
| | Control Rod Hydraulic Return Check Valves (301-95, 98) | 1 | 1 | NA | O | Process |
| | Reactor Head Cooling Check Valves (HS-5) | 1 | | NA | C | Process |
| | Standby Liquid Control Check Valves (SL-7, 8) | 1 | 1 | NA | C | Process |
| 3 | Cleanup Demineralizer System (CU-5) | | 1 | 18 | C | SC |

## 8. Reactor Protection System Response Time

### Description of Changes and Safety Evaluation Summary

The transient analysis for Millstone Unit No. 1 uses a value of 50 msec for the RPS delay time. Section 3.1 of the current Technical Specifications requires that: "The time from initiation of any channel trip to the de-energization of the scram solenoid shall not exceed 100 msec". The GE design specification for the RPS requires a response time of 50 msec. Measurements at plants have demonstrated that the actual response time is less than 50 msec.

If 100 msec were unrealistically used in the transient analysis, the results could be: (a) up to a 0.03 ΔMCPR increase, and (b) an increase in the peak pressures for both the Turbine Generator trips and MSIV flux scram events by approximately 5 psi. The use of 50 msec RPS delay time in the transient analysis is not a safety concern because the actual response time is less than 50 msec.

A review of Millstone Unit No. 1 test data shows the measured time delay in the RPS to be less than 50 msec.

A technical review of this change has found it to be acceptable; a safety evaluation has also been performed in accordance with 10CFR50.59 and has concluded that this change does not constitute an unreviewed safety question. The Millstone Nuclear Review Board has reviewed and approved the proposed change and has concurred with the above determination.

## LIMITING CONDITION FOR OPERATION

### 3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective:

To assure the operability of the reactor protection system.

Specification:

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.

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

## SURVEILLANCE REQUIREMENT

### 4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

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

Specification:

A. Instrumentation systems 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 peak heat flux shall be checked and the APRM scram and rod block settings given by the equations in Specifications 2.1.2A and 2.1.2B shall be determined.

9. Surveillance Testing of ECCS and SLC Equipment

Description of Changes and Safety Evaluation Summary

This Technical Specification change request relates to the Standby Liquid Control System and the ECCS systems (LPCI, Core Spray, Containment Spray, and Emergency Service Water).

As presently stated in the current Millstone Unit No. 1 Technical Specifications, in the event a subsystem or component is out of service, the remaining subsystem or train of that system, and the other core and containment cooling systems, and emergency power sources are all presently required to be tested immediately and daily thereafter.

The Core Spray System testing requires operation of four motor-operated valves including the closure of a normally open admission valve, and the opening and throttling of a test line valve to accommodate a flow of 3,600 gpm. If a core spray subsystem were out of service, this realignment of valves renders the operable subsystem also out of service for the length of the test. In the event of a failure of any component while undergoing tests, the plant is without core spray protection for up to fifteen (15) days, assuming all other components -- LPSI, ESW, diesel generator, and gas turbine generator all function properly.

The LPCI system testing requires the operation of a total of twenty five (25) motor-operated valves (12 in each train + 1 cross connection). One of the normally open admission valves 1-LP-9A or B is closed during pump testing with all flow directed through a three-inch recirculation line. If one LPSI train was out of service, this closure of the admission valve renders the operable subsystem also out of service for the test period. In the event of a failure in operation of one of the twelve motor-operated valves while undergoing this test, the plant would be without LPCI and, therefore, forced to be in cold shutdown within 24 hours.

The emergency service water system consists of two trains -- one for each of the LPCI heat exchangers. There are four pumps and two motor-operated throttling valves. If one pump is out of service, reactor operation is permissible only during the succeeding days provided that during the thirty days all other components of the containment cooling system are operable. This means the flow testing of nine pumps, the operation of thirty motor-operated valves for thirty days with an ever-increasing chance of component failure due to the increased testing.

The Standby Liquid Control System must be available at all times except shutdown. In the event of one SBLC train out of service, the remaining train is tested using demineralized water from a test tank to the SBL control tank. This involves closing the normally locked open suction valve and pumping demineralized water into the SBL control tank; thereby reducing the boron concentration normally required for safety purposes. This valve lineup and subsequent loss of required boron concentration would render the Standby Liquid Control Subsystem out of service for some time.

The requested revisions would eliminate the present requirements to test the remaining train(s) of the ECCS and SLC systems when one train has a component out of service. The changes are technically sound in that there is always at least one train in the proper lineup to perform its design function. Valve and pump operation in the test modes may defeat the system's ability to perform as is, in effect, a reduction of plant safety margin, compared to the proposed changes.

These proposed changes would make the Millstone Unit No. 1 Technical Specifications consistent with the BWR Standardized Technical Specifications for ECCS and SLC surveillance testing.

A technical review of these changes 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.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| | replacement charges to be installed will be selected from the same batch as those tested. Both systems shall be tested and inspected, including each explosion valve in the course of two operating cycles. |
| | b. Manually initiate each system, except the explosion valve and pump solution in the recirculation path back to the storage tank.* |
| | c. Test that the setting of the system pressure relief valves is between 1350 and 1450 psig. |
| B. Operation with Inoperable Components | |
| From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A shall be considered fulfilled, provided that: | |
| 1. The component is returned to an operable condition within 7 days or | |
| 2. A written report shall be submitted to the Atomic Energy Commission when the maintenance to restore the component to an operable condition will last longer than 7 days. | |

Per errata sheet dtd 10-7-70*

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the emergency cooling subsystems.

Objective:

To assure adequate cooling capability for heat removal in the event of a loss of coolant accident or isolation from the normal reactor heat sink.

Specification:

A. Core Spray and LPCI Subsystems

1. Except as specified in 3.5.A.2, 3.5.A.3, 3.5.F.6, 3.5.F.7 and 3.5.F.8, both core spray subsystems shall be operable whenever irradiated fuel is in the reactor vessel.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to periodic testing of the emergency cooling subsystems.

Objective:

To verify the operability of the core and containment cooling subsystems.

Specification:

A. Surveillance of the Core Spray and LPCI Subsystems shall be performed as follows:

1. Core Spray Subsystem Testing:

| Item | Frequency |
|---|---|
| a. Simulated Automatic Actuation Test | Each Refueling Outage |
| b. Pump and Valve Operability | Per Surveillance Requirement 4.13 |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding fifteen days unless such subsystem is sooner made operable, provided that during such fifteen days all active components of the other core spray subsystem and the LPCI subsystem and both emergency power sources required for operation of such components if no external source of power were available shall be operable. | c. Core Spray header $\Delta p$ instrumentation<br>check — Once/day<br>calibrate — Once/3 months<br>test — Once/3 months |
| 3. From and after the date that both core spray subsystems are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless at least one of such subsystems is sooner made operable, provided that during such seven days all active components of the LPCI subsystem and both emergency power sources required for operation of such components if no external source of power were available shall be operable. | |
| 4. Except as specified in 3.5.A.5, 3.5.B.3,4,5, 3.5.F.6, 3.5.F.7 and 3.5.F.8, the LPCI subsystem shall be operable whenever irradiated fuel is in the reactor vessel. | 2. LPCI Subsystem Testing shall be as specified in 4.5.A.1.a, b and c except that three LPCI pumps shall deliver at least 15,000 gpm against a system head corresponding to a reactor vessel pressure of $\geq$ 14.7 psia. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

5. From and after the date that one of the LPCI pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 30 days unless such pump is sooner made operable, provided that during such thirty days the remaining active components of the LPCI and containment cooling subsystem and all active components of both core spray subsystems and both emergency power sources required for operation of such components if no external source of pow were available shall be operable.

6. A maximum of one drywell spray loop may be inoperable for 30 days when reactor water temperature is greater than 212°F.

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

B. Containment Cooling Subsystems

1. Except as specified in 3.5.B.2, 3.5.B.3, 3.5.F.6, 3.5.F.7 and 3.5.F.8, both containment cooling subsystems shall be operable whenever irradiated fuel is in the reactor vessel.

3. During each five-year period, an air test shall be performed on the drywell spray headers and nozzles.

B. Surveillance of the Containment Cooling Subsystems shall be performed as follows:

1. Emergency Service Water Subsystem Testing:

| Item | Frequency |
|---|---|
| a. Pump & Valve Operability | Per Surveillance Requirement 4.13 |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 2. From and after the date that one of the emergency service water (ESW) subsystem pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless pump is sooner made operable, provided that during such thirty days all other active components of the containment cooling system are operable.<br><br>3. From and after the date that one active component in each containment cooling subsystem or a LPCI and ESW in one containment cooling subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days provided the remaining active components in each containment cooling subsystem, both core spray subsystems and both emergency power sources for operation of such components if no external source of power were available shall be operable.<br><br>4. From and after the date that one LPCI and one ESW pump in each containment cooling subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding four days provided the remaining active components of the containment cooling subsystems, both core spray subsystems and both emergency power sources for operation of such components if no external source of power were available, shall be operable. | |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

5. From and after the date that one containment cooling subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding four days provided that all active components of the other containment cooling subsystem, both core spray subsystems and both emergency power sources for operation of such components if no external source of power were available, shall be operable.

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

C. FWCI Subsystem

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

2. There shall be a minimum of 225,000 gallons of water in the condensate storage tank for operation of the FWCI.

C. Surveillance of FWCI Subsystem shall be performed as follows:

1.

| | Item | Frequency |
|---|---|---|
| a. | Pump and valve operability | Per Surveillance Requirement 4.13 |
| b. | Simulated Automatic Actuation Test | Every refueling outage |

2. Once a week the quantity of water in the condensate storage tank shall be logged.

| 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.<br><br>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.<br><br>D. <u>Automatic Pressure Relief (APR) Sybsystems</u><br><br>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. | D. Surveillance of the Automic Pressure Relief Subsystem shall be performed as follows:<br><br>1. During each operating cycle, the following shall be performed:<br><br>a. A simulated automatic initiation of the system throughout its operating sequence but excludes actual valve opening, and<br><br>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. |

Amendment No. 29, 47, 61

LIMITING CONDITION FOR OPERATION

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.



SURVEILLANCE REQUIREMENT

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

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

1. Isolation Condenser System Testing:

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

2. From and after the date that the Isolation Condenser is made or found to be inoperable, for any reason, power operation shall be restricted to a maximum of 40% of full power, i.e., (804 $MW_{th}$) within 24 hours until such time the Isolation Condenser is returned to service.

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

F. <u>Minimum Core and Containment Cooling System Availability</u>

1. Except as specified in 3.5.F.2, 3.5.F.3, 3.5.F.7 and 3.5.F.8 below, both emergency power sources shall be operable whenever irradiated fuel is in the reactor.

POOR ORIGINAL

b. Simulated automatic actuation and functional system testing shall be performed during each refueling outage or whenever major repairs are completed on the system.

c. The system heat removal capability shall be determined once every five years.

d. Calibrate vent line radiation monitors quarterly.

e. Motor operated valves shall be tested per surveillance requirement 4.13.

F. <u>Surveillance of Core and Containment Cooling System</u>

1. The surveillance requirements for normal operation are in Section 3.9.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 2. From and after the date that the diesel generator is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days provided that the gas turbine generator, FWCI, Automatic Pressure Relief Subsystem, all components of the low pressure core cooling and the containment cooling subsystems shall be operable.<br><br>3. From and after the date that the gas turbine generator is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding four days provided that the diesel generator, all components of the APR subsystem, all components of the low pressure core cooling and containment cooling subsystems shall be operable.<br><br>4. If the requirements of 3.5.F.1 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.<br><br>5. Any combination of inoperable components in the core and containment cooling systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions. | |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE PEQUIREMENT |
|---|---|

3.13 INSERVICE INSPECTION

Applicability

Applies to ASME Boiler and Pressure Vessel Code Section III Class 1, 2, and 3 equivalent components.

Objective

To assure the structural integrity of the applicable components defined above.

Specification

The structural integrity of ASME Code Class 1, 2, and 3 equivalent components shall be maintained at an acceptable level in accordance with 10CFR50.55a(g).

4.13 INSERVICE INSPECTION

Applicability

Applies to the periodic inservice inspection and testing of ASME Boiler and Pressure Vessel Code Section III Class 1, 2, and 3 equivalent components.

Objective

To verify the structural integrity of the applicable components defined above.

Specification

A. Inspections

Inservice inspection of ASME Boiler and Pressure Vessel Code Section III Class 1, Class 2, and Class 3 equivalent components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g) with the exemptions and alternate inspections that have been approved by the NRC pursuant to 10CFR50, Section 50.55 a(g)(6)(i). These exemptions and alternate inspections are included in the Inservice Inspection Program.

B. Testing

1. Surveillance with operable components

Inservice testing of ASME Boiler and Pressure Vessel Code Section III Class 1, Class 2, and

3.4 Bases:

A. The design objective of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of 600 ppm of boron in the reactor core in less than 125 minutes. The 600 ppm concentration in the reactor core is required to bring the reactor from full power to a 3% delta k subcritical condition considering the hot and cold reactivity swing and xenon poisoning.

An additional 25% of boron solution is provided for possible imperfect mixing of the chemical solution in the reactor coolant. A minimum quantity of 2720 net gallons of solution having a 13.4% sodium pentaborate concentration is required to meet this shutdown requirement. Actual system volume for this quantity is 2960 gallons. (240 gallons are contained below the pump suction and, therefore, cannot be inserted.)

The time requirement (125 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For the minimum required pumping rate of 32 gallons per minute, the maximum storage volume of the boron solution is established as 4190 gallons.

Boron concentration, solution temperature, (within the tank and connecting piping including check of tank heater and pipe heat tracing system) and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability indicates that monthly testing is adequate to detect if failures have occurred.

Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once during each operating cycle unnecessary. A test of one installed explosive charge is made at least once during each operating cycle to assure that the charges are satisfactory. The replacement charge will be selected from the same batch as the tested charge. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

The relief valves in the standby liquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psi from overpressure. The pressure relief valves discharge back to the standby liquid control solution tank.

B. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made.

3.5 Bases

A. Core Spray and LPCI

This specification assures that adequate emergency cooling capability is available.

Based on the loss of coolant analysis included in Section VI FSAR, either of the two core spray subsystems provides sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature (around 2000°F) 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%. Core spray distribution has been shown, in full scale tests of systems similar in design to that of Millstone Unit 1, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor before the pressure has fallen to 90 psig.

The LPCI subsystem is designed to provide emergency cooling to the core by flooding in the event of a loss of coolant accident. This system is completely independent of the core spray subsystem; however, it does function in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI subsystem in combination with the core spray subsystem provides adequate cooling for break areas of approximately 0.2 square feet up to and including 5.8 square feet, the latter being the double-ended recirculation line break without assistance from the high pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference (1). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems constitute a 1 out of 3 system, however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 45 days and this specification is within this period. For multiple failures, a shorter interval is specified.

Although it is recognized that the information given in reference (1) provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

(1) APED 5736, Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards, April 1969, I. M. Jacobs and P. W. Marriott.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the need for core cooling arise.

Should the loss of one LPCI pump occur, a nearly full complement of core and containment cooling equipment is available. Three LPCI pumps in conjunction with one core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, which will be demonstrated to be operable, a 30-day repair period is justified. If more than one LPCI pump is inoperable, the repair time is set considering the containment cooling function of the LPCI pumps.

B. Containment Cooling Subsystems

The two containment subsystems are provided to remove heat energy from the containment in the event of a loss-of-coolant accident. Each single containment cooling subsystem includes two service water pumps, associated valves, one heat exchanger ($40 \times 10^6$ BTU/hr), two LPCI pumps and necessary instrumentation, control and power equipment. With two heat exchangers (i.e., both loops) operable, it is possible to degrade system performance to one LPCI and one service water pump operating per loop and still not exceed significantly the equipment design temperatures and not rely completely on containment pressure for net positive suction head (NPSH). An interlock to prevent containment spray actuation is included in the design of engineered safety features to prevent inadvertent pressure reduction below that required for NPSH. Reference Amendment Nos. 9, 16, 18, 22 and 23. The heat removal capacity of a single cooling loop is adequate to prevent the torus water temperature from exceeding the equipment temperature capability which is specified to be 203°F in Amendment No. 23. It also provides sufficient subcooling so that adequate NPSH could be assured without reliance on containment pressure except for short intervals during the postulated accident. In the event that only one heat removal loop is operable, station operation will be permitted for four days unless necessary repairs are made to make the other loop operable. A four-day period is selected to permit reasonable time for operator action and maintenance operations.

C. Feedwater Coolant Injection

The feedwater coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or core spray subsystems can protect the core. The FWCI meets this requirement without the use of off-site electrical power. For the pipe breaks for which the FWCI is intended to function, the core never uncovers and is continuously cooled and thus no clad damage occurs. The repair times for the limiting conditions of operation were set considering the use of the FWCI as part of the emergency core cooling system and isolation cooling system.

The FWCI utilizes portions of the normally operating feedwater system; e.g., condensate, condensate booster and feedwater pumps. Therefore, the reliability of the pumps, valves and motors is constantly being demonstrated. Thus the system has an inherently higher degree of reliability than normally non-operating systems. Since an operating string of pump and valves is programmed for FWCI operation, it is not expected that the normally operating portions of the FWCI would be out of operation during normal operation.

D. Automatic Pressure Relief (APR)

The relief valves of the automatic pressure relief subsystem are a back-up to the FWCI subsystem. They enable the core spray or LPCI to provide protection against the small pipe break in the event of FWCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays or LPCI. The core spray and/or LPCI provide sufficient flow of coolant to limit fuel clad temperatures to well below clad melt and to assure that core geometry remains intact.

Redundancy has been provided in the automatic pressure relief function in that only two of the three valves are required to operate. Loss of one of the relief valves does not materially affect the pressure relieving capability and therefore, a 7-day repair period is specified.

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

E. Isolation Condenser System

The turbine main condenser is normally available. The isolation condenser is provided for core decay heat removal following reactor isolation and scram. The isolation condenser has a heat removal capacity sufficient to handle the decay heat production at 300 seconds following a scram. Water will be lost from the reactor vessel through the relief valves in the 300 seconds following isolation and scram. This represents a minor loss relative to the vessel inventory.

4.5 Bases:

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to less than 350 psig, thus, during operation, even if high drywell pressure were simulated, the final valves would not open.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system; i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month.

10. Control Rod Accumulator

Description of Changes and Safety Evaluation Summary

This change would delete the requirement that an inoperable rod accumulator must have no other non-fully inserted control rod with an electrically disarmed control valve, within the nine-rod square array around the affected rod. This deletion does not affect the ability of the rod to scram, and it may be desirable to disarm an adjacent rod, electrically, in some cases.

The intent of the original specification is to ensure an adequate shutdown margin. Since this may be possible via other means, and shutdown margin is governed by other specifications, the deletion would not affect safety.

A technical review of this change has found it to be acceptable; a safety evaluation has also been performed in accordance with 10CFR50.59 and has concluded that this change does not constitute an unreviewed safety question. The Millstone Nuclear Review Board has reviewed and approved the proposed change and has concurred with the above determination.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

| % Inserted From Fully Withdrawn | Average Scram Insertion Times (sec) |
|---|---|
| 5 | 0.398 |
| 20 | 0.954 |
| 50 | 2.120 |
| 90 | 3.800 |

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator.

2. Scram insertion greater than maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.

D. Control Rod Accumulators

Once a shift, check the status in the control room of the pressure and level alarms for each accumulator.

11. Test and Calibration Frequency for Core Cooling Instrumentation

Description of Changes and Safety Evaluation Summary

This proposal involves two changes:

(1) The Instrument Functional Test for the Main Steam Line Isolation Low Pressure instruments will no longer require calibration using simulated electrical signals once every three months, since this is not applicable to pressure switches. The switches are accessible during operation and are calibration-checked each time the mo:.chly system functional t'st is performed. This change is consistent with BWR Standardized Technical Specifications.

(2) The Instrument Functional Test for the Ventilation Exhaust Duct and Refueling Floor Radiation Monitors will be revised. The subject radiation monitors have a built-in test current source which is adequate to functionally test the channel. Since the channels are to be required to be calibrated quarterly (through this change) using a source, it is not reasonable to use a radiation source to functionally test the channel each month, thereby incurring unnecessary radiation exposure to personnel handling the source material.

A technical review of these changes 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.

TABLE 4.2.1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE COOLING INSTRUMENTATION ROD BLOCKS AND ISOLATIONS

| Instrument Channel | Instrument Functional Test (2) | Calibration (2) | Instrument Check (2) |
|---|---|---|---|
| **ECCS Instrumentation** | | | |
| 1. Reactor Low-Low Water Level | (1) | Once/3 Months | - |
| 2. Drywell High Pressure | (1) | Once/3 Months | - |
| 3. Reactor Low Pressure (Pump Start) | (1) | Once/3 Months | - |
| 4. Reactor Low Pressure (Valve Permissive) | (1) | Once/3 Months | - |
| 5. APR LP Core Cooling Pump Interlock | (1) | Once/3 Months | - |
| 6. Containment Spray Interlock | (1) | Once/3 Months | - |
| 7. Loss of Normal Power Relays | Refueling Outage | None | - |
| 8. Power Available Relays | (1) (5) | None | - |
| 9. Reactor High Pressure | | Once/3 Months | - |
| **Rod Blocks** | | | |
| 1. APRM Downscale | (1) (3) | Once/3 Months | (1) |
| 2. APRM Flow Variable | (1) (3) | Once/3 Months | (1) |
| 3. IRM Upscale | (6) | (6) | (6) |
| 4. IRM Downscale | (6) | (6) | (6) |
| 5. RBM Upscale | (1) (3) | Once/3 Months | (1) |
| 6. RBM Downscale | (1) (3) | Once/3 Months | (1) |
| 7. SRM Upscale | (6) | (6) | (6) |
| 8. SRM Detector not in Startup Position | (6) | (6) | (6) |
| **Main Steam Line Isolation** | | | |
| 1. Steam Tunnel High Temperature | Refueling Outage | Refueling Outage | - |
| 2. Steam Line High Flow | (1) | Once/3 Months | Once/Day |
| 3. Steam Line Low Pressure | (1) | Once/3 Months | None |
| 4. Steam Line High Radiation | (1) (3) | Once/3 Months (4) | Once/Day |





Amendment No. ~~24~~, 57

TABLE 4.2.1 (Continued)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE COOLING INSTRUMENTATION ROD BLOCKS AND ISOLATIONS



| nstrument Channel | Instrument Functional Test (2) | Calibration (2) | Instrument Check (2) |
|---|---|---|---|
| solation Condenser Isolation | | | |
| . Steam Line High Flow | (1) | Once/3 Months | (1) |
| . Condensate Line High Flow | (1) | Once/3 Months | (1) |
| eactor Building Ventilation and Standby Gas Treatment System Initiation | | | |
| . Ventilation Exhaust Duct and Refueling Floor Radiation Monitors | (1) (3) | Once/3 Months | Once/Day |
| ir Ejector Off-Gas Isolation | | | |
| . Radiation Monitors | (1) (3) | Once/3 Months (4) | Once/Day |

otes:

1) Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0 X 105, thereafter according to Figure 4.1.1 with an interval not less than one month nor more than three months. Millstone will use data compiled by Commonwealth Edison on the Dresden 2 Unit in addition to Millstone Unit 1 data.

2) Functional test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped.

3) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. See Note 4.

4) These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed during each refueling outage.

5) The individual power available on emergency bus relays will be functionally tested at the frequency specified by (1) above. A full functional test including the actuation of the permissives will be performed every refueling outage.

6) This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations including the sensors will be performed during each refueling outage. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

12. Personnel Air-Lock Door Leak Test

Description of Changes and Safety Evaluation Summary

This change would revise the test pressure and frequency for performing the Personnel Air-Lock Door Local Leak Rate Test. The change is in agreement with the 10CFR50, Appendix J requirements, as documented in the NRC letter (G. Lear) to NNECO (D. C. Switzer) dated March 3, 1977.

A technical review of these changes 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.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

e. Local Leak Rate Tests (LLRT)

(1) Primary containment testable pentrations and isolation valves shall be tested at a pressure of 43 psig except the main steam line isolation valves shall be tested at a pressure of 25 psig each operating cycle. Bolted double-gasketed seals shall be tested whenever the seal is closed after being opened and at least once during each operating cycle.

(2) Personnel air lock door seals shall be tested at a pressure of 43 psig at least once every 6 months. If the airlock is opened when primary containment integrity is required during the interval between the above tests, the air lock door seals shall be tested at 10 psig within 72 hours of the first of a series of opening.

f. Acceptance criteria and corrective action for LLRT:

If the total leakage rates listed below are exceeded, repairs and retests shall be performed to correct the condition.

(1) Double-gasketed seals 10% $L_{to}$ (43).

(2) (a) Testable penetrations and isolation valves 30% $L_{to}$ (43).

(b) Any one penetration or isolation valve except main steam isolation valves 5% $L_{to}$ (43).

The penetration and air purge piping leakage test frequency, along with the containment leak tests, is adequate to allow detection of leakage trends. Whenever a double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determined that the seals are performing properly. The test pressure of 43 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

The results of the loss-of-coolant accident analyses presented in Amendment No. 17 of Dresden Unit 2 (Docket No. 50-237) indicate that fission products would not be released directly to the environs because of leakage from the main steam line isolation valves due to holdup in the steam system complex. Although this affect would indicate that an adequate margin exists with regard to the release of fission products, a program will be undertaken to further reduce the potential for such leakage to bypass the standby gas treatment system.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

Surveillance of the suppression chamber-drywell vacuum breakers consists of operability checks, calibration of instrumentation and inspection of the valves.

The monthly operability tests are performed to check the capability for the disc to open and close and to functionally test the position indication system. This test frequency is justified based on previous experience and the fact that these valves are normally closed and are only open during tests or accident conditions.

The refueling outage surveillance tests are performed to check that the valve will perform properly during the accident condition and to verify the calibration of the position indication system. Measuring the force required to lift the valve assures that the valve will function properly during an accident. Inspection of a select number of valves during each refueling outage assures that deterioration of the valve internals or misalignment of the disc does not impair the proper operation of the valve. This test interval is based on equipment quality and previous equipment experience.

13. Rod Worth Minimizer Diagnostic Test

Description of Changes and Safety Evaluation Summary

This change deletes the Surveillance Requirement to perform the rod worth minimizer computer line diagnostic test. The new rod worth minimizer does not have a diagnostic test, and if the backup rod worth minimizer is used, the diagnostic test is not necessary to prove proper rod worth minimizer operation. The level of protection provided by the rod worth minimizer is not degraded by this change.

A technical review of this change has found it to be acceptable; a safety evaluation has also been performed in accordance with 10CFR50.59 and has concluded that this change does not constitute an unreviewed safety question. The Millstone Nuclear Review Board has reviewed and approved the proposed change and has concurred with the above determination.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

3.3.B Contr 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) Proper annunciation of the select error of at least one out-of-sequence control rod in each fully inserted group shall be verified.

    (iii) 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.

14. Housekeeping Changes

Description of Changes and Safety Evaluation Summary

A number of minor housekeeping changes are proposed as described below, to correct typographical errors, grammatical errors, and outdated specifications.

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.

| Page | Section | Reason |
|---|---|---|
| B 2-8 | Bases 2.1.2 | Correct word error |
| 3/4 2-2 | Table 3.2.1 | Correction of statement |
| 3/4 2-4 | Table 3.2.2 | Information deleted during reprint |
| 3/4 2-8 | 3.2.E.1 | Grammatical correction |
| 3/4 3-2 | 4.3.B.1.b | Grammatical correction |
| 3/4 3-3 | 4.3.B.3.b | To be consistent with analysis and basis |
| 3/4 3-4 | 3.3.C.1 | Reprinted incorrectly |
| 3/4 6-10 | 4.6.H | Title deleted during reprint |
| 3/4 7-14 | 4.7.D.1.a | Misspelled word |
| 3/4 7-15 | Table 3.7.1 | Typing Error (I not 1) and two missing parentheses |
| 3/4 7-17 | 4.7.D.1 | Last "c" should be "d" |
| 3/4 9-2 | 3.9.B.2 | General update (AEC is now NRC) |
| 3/4 9-3 | 3.9.6 | References wrong section |
| 3/4 9-3 | 4.9.B.1.c | Two battery banks |
| 3/4 11-1 | 4.11 | Worded incorrectly after reprint |

| Page | Section | Reason |
|---|---|---|
| 3/4 12-4 | 4.12.A.1.h | Do not have 24 volt battery bank, but 2 12 volt batteries |
| 3/4 12-4 | 4.12.A.1.h.1.a | Statement clarification to comply with design |
| 3/4 12-4 | 4.12.A.1.h.1.b | Statement clarification to comply with design |
| 3/4 12-4 | 4.12.A.1.h.2 | Statement clarification to comply with design |
| 3/4 12-4 | 4.12.A.1.h.3.b | No battery-to-battery connections |
| B 3/4 1-2 | Bases 3.1 | Grammatical correction |
| B 3/4 1-8 | Bases 4.1 | Consistency |
| B 3/4 2-2 | Bases 3.2 | General update (AEC is now NRC) |
| B 3/4 2-2 | Bases 3.2 | This Bases commitment was fulfilled via NNECO letter to NRC (D. L. Ziemann) October 10, 1978. |
| B 3/4 3-1 | Bases 3.3 | Typing error |
| B 3/4 4-1 | Bases .4 | Typing error during reprint |
| B 3/4 5-4 | Bases 3.5 | Statement clarification to comply with present mode of operation |
| B 3/4 6-4 | Bases 3.5 and 4.6 | Nomenclature clarification |
| B 3/4 6-4 | Bases 3.6 and 4.6 | Nomenclature clarification |
| B 3/4 7-2 | Bases 3.7 | Grammatical correction |
| B 3/4 7-3 | Bases 3.7 | Grammatical correction |
| B 3/4 7-8 | Bases 4.7 | General update (AEC is now NRC) |
| B 3/4 7-9 | Bases 4.7 | Typing error |
| B 3/4 7-10 | Bases 4.7 | Typing error |

| Page | Section | Reason |
|---|---|---|
| B 3/4 9-1 | Bases 3.9 | Update to comply with existing station design |
| B 3/4 9-1 | Bases 3.9 | Update to comply with existing station design |
| B 3/4 9-1 | Bases 3.9 | Update to comply with existing station design |
| 5-1 | 5.1 | Update Company name |

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 now 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.

TABLE 3.2.1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION FUNCTIONS

| Minimum Number of Operable Instrument Channels Per Trip System (1) | Instruments | Trip Level Setting | Action (3) |
|---|---|---|---|
| 2 | Reactor Low Water | ≥ 127 inches above top of active fuel | A |
| 2 | Reactor Low Low Water | 79 (+4-0) inches above top of active fuel | A |
| 2 (4) | High Drywell Pressure | ≤ 2 psig | A |
| 2 (2) (5) | High Flow Main Steamline | ≤ 120% of rated steam flow | B |
| 2 of 4 in each of 2 subchannels | High Temperature Main Steamline Tunnel | ≤ 200°F | B |
| 2 | High Radiation Main Steamline Tunnel | ≤ 7 times normal rated power background | B |
| 2 | Low Pressure Main Steamlines | ≥ 880 psig | B |
| 2 | High Flow Isolation Condenser Line | 164 inches ≥ trip setting (water differential on steam line) ≥ 150 inches. 44 inches ≥ trip setting (water differential on water side) ≥ 35 inches. | C |

(1) Whenever primary containment integrity is required, there shall be two operable or tripped trip systems for each function, except for low pressure main steamline which only need be available in the Run position.

(2) Per each steamline.

(3) Action: If the first column cannot be met for one of the trip systems, that trip system shall be tripped. If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

A. Initiate an orderly shutdown and have reactor in cold shutdown condition in 24 hours.
B. Initiate an orderly load reduction and have reactor in Hot Standby within 8 hours.
C. Close isolation valves in isolation condenser system.

(4) May be bypassed when necessary by closing the manual instrument isolation valve for PS-1621, A through D, during purging for containment inerting or deinerting. |

(5) Minimum number of operable instrument channels per trip system requirement does not have to be met for a steamline if both containment isolation valves in the line are closed.

TABLE 3.2.2 (Continued)

INSTRUMENTATION THAT INITIATES AND CONTROLS THE EMERGENCY CORE COOLING SYSTEMS

| Minimum Number of Operable Inst. Channels Per Trip System (1) | Trip Function | Trip Level Setting | Remarks |
|---|---|---|---|
| 1 | Timer Auto Blowdown | $\leq$ 120 seconds | 1 - In conjunction with low low reactor water level and high drywell pressure and LP core cooling pump interlock. |
| 2 | Containment Spray Interlock | $\geq$ 2/3 Core Height i.e., $\geq$ 166 inches<br>4.5 $\leq$ P $\leq$ 5.5 Psig | 1 - Prevent inadvertent operation of containment spray. |
| 2 | APR LP Core Cooling Pump Interlock | 90 $\leq$ P $\leq$ 110 Psig | 1 - Defer APR actuation pending confirmation of LP core cooling system operation. |
| 2 sets of 2 (Total) | Power Available on Emergency Buses | Normal for 120 Volt undervoltage relays (Monitor Buses 5 and 6) | 1 - Permissive for auto-close of emergency power source on buses.<br>2 - Permissive to start core spray and LPCI pumps. |
| 2 sets of 7 (Total) | Loss of Normal Power | Normal for 120 Volt Undervoltage Relays (Monitor Buses 1,2,3 and 4) | 1 - Initiates start of emergency power sources.<br>2 - Strip loads from buses.<br>3 - Permissive for emergency power sources to close on buses. |

1) If the first column cannot be met for one of the trip systems, that system may be tripped. If the first column cannot be met for both trip systems, immediately initiate an orderly shutdown to cold conditions.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
| --- | --- |
| 2. The minimum number of operable instrument channels specified in Table 3.2.3 for the Rod Block Monitor may be reduced by one for maintenance and/or testing for periods not in excess of 24 hours in any 30-day period. | |
| D. <u>Air Ejector Off-Gas System</u> | |
| 1. Except as specified in 3.2.D.2 below, both air ejector off-gas system radiation monitors shall be operable during reactor power operation. The trip settings for the monitors shall be set at a value not to exceed the equivalent of the instantaneous stack release limit specified in Specification 3.8. The time delay setting for closure of the steam jet-air ejector off-gas isolation valve shall not exceed 15 minutes. | |
| 2. From and after the date that one of the two air ejector off-gas system radiation monitors is made or found to be inoperable, reactor power operation is permissible only during the succeeding 24 hours, provided the inoperable monitor is tripped, unless such system is sooner made operable. | |
| E. <u>Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation</u> | |
| 1. Except as specified in 3.2.E.2 below, four radiation monitors shall be operable at all times. | |

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 not 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 ne[illegible] 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 d[illegible] 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.

## LIMITING CONDITION FOR OPERATION

3.3.B. Control Rod Withdrawal

3. (a) Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would not make the core more than 0.013 delta k supercritical.

(b) 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. After October 1, 1974, 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.

## SURVEILLANCE REQUIREMENT

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 computer on-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 20% 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 |
|---|---|

5. During operation with limiting control rod patterns, as determined by the reactor engineer, either:

   a. Both RBM channels shall be operable; or

   b. Control rod withdrawal shall be blocked; or

   c. The operating power level shall be limited so that the MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

| % Inserted From Fully Withdrawn | Average Scram Insertion Times (sec) |
|---|---|
| 5 | 0.375 |
| 20 | 0.900 |
| 50 | 2.000 |
| 90 | 3.500 |

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

C. Scram Insertion Times

During each operating cycle, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished during reactor power operation, the measured scram insertion times shall be extrapolated to the reactor power operation condition utilizing previously determined correlations.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

G. Jet Pumps

1. Whenever the reactor is in the startup/Hot Standby or Run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication cannot be obtained for at least nineteen jet pumps, an orderly shutdown shall be initiated within 12 hours and the reactor shall be in a cold shutdown condition within 24 hours.

H. Recirculation Pump Flow Mismatch

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

G. Jet Pumps

1. Whenever there is a recirculation flow with the reactor in the startup/hot standby or run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.

3. The baseline data required to evaluate the conditions in Specifications 4.6.G.1 and 4.6.G.2 will be acquired each operating cycle.

H. Recirculation Pump Flow Mismatch

1. Recirculation pump speed shall be checked daily for mismatch.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
| --- | --- |

d. The fuel cask or irradiated fuel is not being moved within the reactor building.

D. Primary Containment Isolation Valves

1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.

capability shall be demonstrated at three or more points within the containment prior to fuel movement and may be demonstrated up to 10 days prior to fuel movement. Secondary containment capability need not be demonstrated more than once per operating cycle unless damage or modifications to the second ry containment have violated the integrity of the pressure retaining boundary of that structure.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

   a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

   b. At least once per operating cycle the instrument line flow check valves shall be tested for proper operation.

   c. At least once per quarter:

      1) All normally open power-operated isolation valves (except for the main steam-line power-operated isolation valves) shall be fully closed and reopened.

TABLE 3.7.1

PRIMARY CONTAINMENT ISOLATION

| Isolation Group | Valve Identification (Valve Number) | Number of Power Operated Valves Inboard | Outboard | Maximum Operating Time (Sec) | Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| 1 | Main Steam Line Isolation (MS-1A, 2A, 1B, 2B, 1C, 2C, 1D, 2D) | 4 | 4 | $3 \leq T \leq 5$ | 0 | GC |
| 1 | Main Steam Line Drain (MS-5) | 1 | | 35 | C | SC |
| 1 | Main Steam Line Drain (MS-6) | | 1 | 35 | C | SC |
| 1 | Recirculation Loop Sample Line (SM-1, 2) | 1 | 1 | 5 | C | SC |
| 1 | Isolation Condenser Vent to Main Steam Line (IC-6, 7) | | 2 | 5 | 0 | GC |
| 2 | Drywell Floor Drain (SS-3, 4) | | 2 | 20 | 0 | GC |
| 2 | Drywell Equipment Drain (SS-13, 14) | | 2 | 20 | 0 | GC |
| 2 | Drywell Vent (AC-7) | | 1 | 10 | C | SC |
| 2 | Drywell Vent Relief (AC-9) | | 1 | 15 | C | SC |
| 2 | Drywell and Suppression Chamber Vent from Reactor Building (AC-8) | | 1 | 10 | C | SC |
| 2 | Drywell Vent to Standby Gas Treatment System (AC-10) | | 1 | 10 | C | SC |
| 2 | Suppression Chamber Vent (AC-11) | | 1 | 10 | C | SC |
| 2 | Suppression Chamber Vent Relief (AC-12) | | 1 | 15 | C | SC |
| 2 | Suppression Chamber Supply (AC-6) | | 1 | 10 | C | SC |
| 2 | Drywell Supply (AC-5) | | 1 | 10 | C | SC |
| 2 | Drywell and Suppression Chamber Supply (AC-4) | | 1 | 10 | C | SC |
| 3 | Cleanup Demineralizer System (CU-2) | 1 | | 18 | 0 | GC |
| 3 | Cleanup Demineralizer System (CU-3, 5, 28) | | 3 | 18 | 0 | GC |
| 3 | Shutdown Cooling System (SD-1) | 1 | | 48 | C | SC |
| 3 | Shutdown Cooling System (SD-2A, 2B, 4A, 4B) | | 4 | 48 | C | SC |
| 3 | Shutdown Cooling System (SD-5) | | 1 | 48 | C | SC |
| 3 | Reactor Head Cooling Line (HS-4) | | 1 | 45 | C | SC |
| 4 | Isolation Condenser Steam Supply (IC-1) | 1 | | 24 | 0 | GC |
| 4 | Isolation Condenser Steam Supply (IC-2) | | 1 | 24 | 0 | GC |
| 4 | Isolation Condenser Condensate Return (IC-3) | 1 | | 19 | 0 | GC |
| 4 | Isolation Condenser Condensate Return (IC-4) | | 1 | 19 | C | SC |
| | Feedwater Check Valves (FW-9A, 10A, 9B, 10B) | 2 | 2 | NA | 0 | Process |
| | Control Rod Hydraulic Return Check Valves (301-95, 98) | 1 | 1 | NA | 0 | Process |
| | Reactor Head Cooling Check Valves (HS-5) | 1 | | NA | C | Process |
| | Standby Liquid Control Check Valves (SL-7, 8) | 1 | 1 | NA | C | Process |

Amendment No.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

2. In the event any isolation valve specified in Table 3.7.1 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specification 3.7.D cannot be met, initiate an orderly shutdown and have reactor in the cold shutdown condition within 24 hours.

2) With the reactor power less than 75% of rated, trip main steam isolation valves (one at a time) and verify closure time.

d. At least once per month, the main steam-line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

2. Whenever an isolation valve listed in Table 3.7.1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
| --- | --- |
| | |

5. All station and switchyard 24 and 125 volt batteries and associated battery chargers are operable.

B. When the mode switch is in Run, the availability of power shall be as specified in 3.9.A, except as specified below:

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.

2. From the 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.

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.

c. During the monthly generator test, the diesel fuel oil transfer pumps shall be operated.

2. Gas Turbine Generator

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.

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.

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 |
|---|---|
| 4. From and after the date that one of the two 125 volt or 24 volt battery systems is made or found to be inoperable for any reason reactor operation is permissible only during the succeeding seven days unless such battery system is sooner made operable. | c. At every refueling outage or at 18 months intervals, the station batteries shall be subjected to a performance test in accordance with the procedures described in Section 5.4 in IEEE Standard 450-1972, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Stationary Type Power Plant and Substation Lead Storage Batteries". |
| C. Diesel and Gas Turbine Fuel | 2. Switchyard Batteries |
| There shall be a minimum of 20,000 gallons of diesel fuel supply onsite for the diesel and a minimum of 35,000 gallons onsite for the gas turbine, except as permitted in Specification 3.9.B.2 | 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. |
| | C. The quantity of gas turbine generator and diesel generator fuel shall be logged weekly and after each operation of the unit. |
| | Once a month a sample of the diesel and gas turbine fuel shall be taken from the underground storage tanks and checked for quality. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|

3.11 REACTOR FUEL ASSEMBLY

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

1. During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed 91% of the limiting value shown in Figure 3.11.1. This limiting value of APLHGR has been provided to insure fuel cladding integrity under postulated small break LOCA conditions with a gas turbine failure and LPCI injection into the broken loop.

2. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR specified in Section 3.11.A.1 is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.11 REACTOR FUEL ASSEMBLY

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥ 25% rated thermal power.



Amendment No. ~~1~~, ~~16~~, ~~28~~, ~~34~~, ~~46~~, ~~61~~

(b) Verifying the diesel starts from ambient conditions on the auto-start signal and operates for $\geq 20$ minutes while loaded with the fire pump.

h. The fire pump diesel starting 12-volt batteries and charger shall be demonstrated OPERABLE:

1. At least once per 7 days by verifying that:

   (a) The electrolyte level of each battery cell is above the plates, and

   (b) The individual overall battery voltages are $\geq$ 12 volts.

2. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the batteries.

3. At least once per 18 months by verifying that:

   (a) The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and

   (b) The terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

APRM's #4, #5 and #6 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water, generator load rejection, and turbine stop valve closure are discussed in Section 2 of these specifications.

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

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. A part of this piping is an instrument volume which accommodates in excess of 39 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrumented volume which alarm and scram the reactor when the volume of water reaches 39 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

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

the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument. While included in Group (C), the Condenser Low Vacuum trip is treated differently. This is because the condenser low vacuum trip sensor can only be tested during shutdown. The primary function of this trip is to protect the turbine and condenser, although it is connected into the reactor protection system; thus testing the sensor at each refueling outage is adequate.

Calibration frequency of the instrument channels is divided into two groups. These are as follows:

a. Passive type indicating devices that can be compared with like units on a continuous basis.

b. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a drift of 0.4% would occur and thus providing for adequate margin. For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

B. The peak heat flux shall be checked once per day to determine if the APRM scram requires adjustment. This will normally be done by checking the LPRM readings. Only a small number of control rods are moved daily and thus the peaking factors are not expected to change significantly and thus a daily check of the peak heat flux is adequate.

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge and sump isolation valves, and reactor building ventilation isolation valves. Group 2 actuation also initiates the SBGTS. High drywell pressure activates only these valves because high drywell pressure could occur as the result of non-safety related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The low low water level instrumentation initiates protection for the full spectrum of loss of coolant accidents and causes a trip of all primary system isolation valves.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring the steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 120% of rated steam flow in conjunction with the flow limiters and main steamline closure, limit the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500°F and release of radioactivity to the environs is well below 10 CFR 100 guideline values. The main steamline high flow break detection is a one out of two twice logic for each individual steamline, four detectors per line for a total of 16 detectors. When a steamline is isolated by closing both main steam isolation valves the operable instrument channels per trip system requirements are not required to be met because the protection afforded by the remaining operable logic in the in-service steamlines provides complete recognition of the steam flow measurements required for correct protective action.

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

High radiation monitors in the main steamline tunnel have been provided to detect gross fuel failure. This instrumentation causes closure of Group 1 valves, the only valves required to close to prevent further release to the environment. With the established setting of seven times normal background, and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guideline values are not exceeded for the most rapid failure mechanism postulated (control rod drop accident).

Pressure instrumentation is provided which trips when main steamline pressure at the turbine drops below 880 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "Refuel," "Shutdown," and "Startup/Hot Standby" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 880 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there is no release of fission products other than those in the reactor water.

3.3 Bases:

A. Reactivity Limitations

The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. The reactivity of the loaded core will be limited so the core can be made subcritical by at least R + 0.33% $\Delta K^{+}$ in the most reactive condition during the operating cycle, with the strongest control rod fully withdrawn and all others fully inserted. The value of R in %$\Delta$K is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check, i.e., the initial loading. R must be a positive quantity or zero. The value of .08% $\Delta$K has been added to the normal value of 0.25% $\Delta$K to allow for potential maximum settling of the $B_4C$ powder in the inverted control rod tubes still remaining in the core. A core which contains temporary control or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum and then decreases thereafter. See Figure 3.3.2 of the FSAR for such a curve.

The value of R is the difference between the calculated core reactivity at the beginning of the operating cycle and the calculated value of core reactivity any time later in the cycle where it would be greater than at the beginning. For the first fuel cycle, R was calculated to be not greater than 0.10% $\Delta$K. A new value of R must be determined for each fuel cycle.

The 0.33% $\Delta$K in the expression R + 0.33% $\Delta$K is provided as a finite, demonstrable, subcriticality margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.33% $\Delta$K in reactivity. Observation of subcriticality in this condition assures subcriticality with not only the strongest rod fully withdrawn but at least a R + 0.33% $\Delta$K margin beyond this.

2. Reactivity Margin - Stuck Control Rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If the rod is fully inserted and then disarmed electrically,* it is in a safe position of maximum contribution to shutdown reactivity. If it is disarmed elctrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shutdown at all times with the remaining control rods assuming the strongest operable control rod does not insert. An allowable pattern for control rods out of service, which shall meet this specification, will be available to the operator.

---

*To disarm the drive electrically, four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the drive immoveable. This procedure is equivalent to valving out the drive and is preferred because, in this condition, drive water cools and minimizes crud accumulation in the drive.

+See October 22, 1974 Inverted Control Rod Inspection and Analysis Report, and Technical Specifications Change Request.

3.4 Bases:

A. The design objective of the liquid control system is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of 600 ppm of boron in the reactor core in less than 125 minutes. The 600 ppm concentration in the reactor core is required to bring the reactor from full power to a 3% delta k subcritical condition considering the hot to cold reactivity swing and xenon poisoning.

An additional 25% of boron solution is provided for possible imperfect mixing of the chemical solution in the reactor coolant. A minimum quantity of 2720 net gallons of solution having a 13.4% sodium pentaborate concentration is required to meet this shutdown requirement. Actual system volume for this quantity is 2960 gallons. (240 gallons are contained below the pump suction and, therefore, cannot be inserted.)

The time requirement (125 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For the minimum required pumping rate of 32 gallons per minute, the maximum storage volume of the boron solution is established as 4190 gallons.

Boron concentration, solution temperature, (within the tank and connecting piping including check of tank heater and pipe heat tracing system) and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability indicates that monthly testing is adequate to detect if failures have occurred.

Components of the system are checked periodically as described above and make a functional test of the entire system on a frequency of less than once during each operating cycle unnecessary. A test of one installed explosive charge is made at least once during each operating cycle to assure that the charges are satisfactory. The replacement charge will be selected from the same batch as the tested charge. A continual check of the firing circuit continuity is provided by pilot lights in the control room.

The relief valves in the standby .iquid control system protect the system piping and positive displacement pumps which are nominally designed for 1500 psi from overpressure. The pressure relief valves discharge back to the standby liquid control solution tank.

B. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the remaining system will perform its intended function and that the reliability of the system is good is obtained by demonstrating operation of the pump in the operable circuit at least once daily.

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1085 psig sustained for 15 seconds. The time delay is provided to prevent unnecessary actuation of the system during turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. Make-up water to the shell side of the isolation condenser can be provided by the condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are operable from on-site power. The fire protection system is also available as make-up water. An alternate method of cooling the core upon isolation from the main condenser is by using the relief valves and FWCI subsystem in a feed and bleed manner. The minimum shell side water volume in the isolation condenser is 15,500 gallons.

The function of the Isolation Condenser during a small break accident is to assist the automatic pressure relief system in depressurizing the reactor as a backup to the FWCI system. The two effects of isolation condenser depressurization are: (1) the minimization of reactor inventory loss which normally occurs during APR blowdown; this reduces the time of core uncovery prior to reflooding; and (2) earlier onset of low pressure core spray cooling.

Analysis performed by General Electric in March 1976, in support of extended operation of Millstone while the isolation condenser was being retubed indicated that from 40% rated power, over 30 minutes is available to initiate operator action to mitigate the consequences of a loss of all feedwater. This is based upon manual depressurization with APR and coolant supplied by the LPCI and core spray systems. The FWCI was assumed lost as part of the non-mechanistic assumption of loss of feedwater. The successful mitigation of this postulated event was no uncovering of the fuel. Operators are instructed regarding special procedures to be utilized during this mode of plant operation. Thus, reducing power to 40% when the isolation condenser is inoperable provides a limiting condition for operation that is sufficient to preclude the need for any additional limiting conditions for operation on other ECCS systems.

## F. Emergency Cooling Availability

The purpose of Specification F is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the emergency power source which powered the opposite core spray were out of service, only two LPCI pumps would be available. Likewise, if two LPCI pumps were out of service and two emergency service water pumps on the opposite side were also out of service, no containment cooling would be available. It is during refueling outages that major maintenance is performed and during such time that low pressure core cooling systems may be out of service depending on the activities being performed. Specificification F allows removal of one CRD mechanism or fuel removal and replacement while the torus is in a drained condition without compromising core cooling capability. The specification establishes the minimum operable low pressure core cooling system, water inventories, electrical power supplies and other additional requirements that must exist to allow such activities as CRD mechanism maintenance or fuel removal and replacement. to be performed in parallel with other major activities. The available core cooling capability for a potential draining of the reactor vessel while this work is performed is based on an estimated drain rate and the maintained minimum volume of water, 383,000 gallons, in the refueling cavity to be supplied to the reactor

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 floor drain sump pumps is 50 gpm and the capacity of either of the drywell equipment drain sump 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 open. 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.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally change very slowly and monitoring these parameters once per shift is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

3. and 4. Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber; and suppression chamber and reactor building so that the structural integrity of the containment is maintained.

The vacuum relief system between the pressure suppression chamber and reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure defferential less than 1 psig; the external design pressure. One valve may be out of service for repairs for a period of seven days. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The design basis loss of coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5%/day at 43 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 5 rem and the maximum total thyroid dose is about 125 rem at the site boundary over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population distance of 2.3 miles are 4 rem whole body and 155 rem maximum total thyroid dose. Thus, these doses reported are the maximum that would be expected in the unlikely event of a design basis loss of coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines. The fission product source term defined in TID-14844 was also used in the design of facility engineered safety features including shielding and filter sizing.

The maximum allowable test leak rate is 1.2%/day at a pressure of 43 psig. This value for the test condition was derived from the maximum allowable accident leak rate of about 1.5%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air depleted of oxygen whereas under test conditions the test atmosphere would be air or nitrogen at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 and determined from the guide on containment testing.[1]

Although the does calculations suggest that the accident leak rate could be allowed to increase to about 3.0%/day before the guidelines thyroid dose value given in 10 CFR 100 would be exceeded, establishing the test limit of 1.2%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The operational limit is derived by multiplying the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the NRC guide for developing leak rate testing and surveillance of reactor containment vessels.[2] Allowing the test intervals to be extended up to 8 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

(1) TID 20583, Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations.
(2) Technical Safety Guide, Reactor Containment Leakage Testing and Surveillance Requirements, USAEC, Division of Safety Standards, Revised Draft, December 15, 1966.

The penetration and air purge piping leakage test frequency, along with the containment leak tests, is adequate to allow detection of leakage trends. Whenever a double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determined that the seals are performing properly. The test pressure of 43 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

The results of the loss-of-coolant accident analyses presented in Amendment No. 17 of Dresden Unit 2 (Docket No. 50-237) indicate that fission products would not be released directly to the environs because of leakage from the main steam line isolation valves due to holdup in the steam system complex. Although this affect would indicate that an adequate margin exists with regard to the release of fission products, a program will be undertaken to further reduce the potential for such leakage to bypass the standby gas treatment system.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

Surveillance of the suppression chamber-drywell vacuum breakers consists of operability checks, calibration of instrumentation and inspection of the valves.

The monthly operability tests are performed to check the capability for the disc to open and close and to functionally test the position indication system. This test frequency is justified based on previous experience and the fact that these valves are normally closed and are only open during tests or accident conditions.

The refueling outage surveillance tests are performed to check that the valve will perform properly during the accident condition and to verify the calibration of the position indication system. Measuring the force required to lift the valve assures that the valve will function properly during an accident. Inspection of a select number of valves during each refueling outage assures that deterioration of the valve internals or misalignment of the disc does not impair the proper operation of the valve. This test interval is based on equipment quality and previous equipment experience.

B. Standby Gas Treatment System and

C. Secondary Containment

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 7 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow RDT Standard M-16-1T. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality.

Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N101.1-1972. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52. Although the SGTS design flow rate is 1100 SCFM, the DOP test at reduced flow rate is actually more sensitive because diffusion is the primary mechanism of small particle collection. The lower limit for test flow rate (500 SCFM) is based on test instrument sensitivity.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of the heaters will prevent moisture buildup in the filters and adsorber system.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other system must be tested daily. This substantiates the availability of the operable system and thus reactor operation may continue.

3.9 Bases:

A. The objective of the auxiliary electric power availability specification is to assure that adequate power will be available to operate the emergency safeguards equipment. Adequate power can be provided by any one of the following power sources: one 345 kv line, the 27.6 kv system, the gas turbine-generator and the diesel generator.

This specification assures that at least two offsite and two onsite power sources will be available before the reactor is started up. In addition to assuring power source operability, all of the associated switchgear and vital equipment must be operable as specified to assure that the emergency cooling equipment can be operated, if required, from the power sources.

B. Normally, three 345 kv lines will be available to provide emergency power to the plant when the reactor is operating. However, adequate power is available with only one 345 kv line in service. Therefore, reactor operation is permitted for up to seven days with only one 345 kv line in service to accommodate necessary maintenance, etc.

In the event that all 345 kv lines are out of service, continued reactor operation is permitted provided both onsite emergency power sources are operating with an adequate fuel supply. Two operational power sources provide an adequate assurance of emergency power availability under these circumstances. In addition, the isolation condenser system is required to be operable as a standby heat removal system.

Normally both the gas turbine generator and diesel generator are required to be operable to assure adequate emergency power with no offsite power sources. However, due to the redundancy and reliability of offsite power, one of the two emergency onsite power sources may be out of service for limited periods of time providing two offsite power sources are available during these periods.

C. Either of the two station batteries has enough capability to energize the vital buses and power the other emergency equipment. Due to the high reliability of battery systems, one of the two batteries may be out of service for up to 7 days. This minimizes the probability of unwarranted shutdowns by providing adequate time for reasonable repairs.

D. The diesel fuel supply of 20,000 gallons will supply the diesel generator with about five days of full load operation. The gas turbine generator fuel supply of 35,000 gallons is sufficient to operate the unit for at least two and one-half days considering the fuel consumption vs. load and load vs. time requirements during the postulated accident. Reference Amendment 18. Additional fuel can be supplied to the site within twelve hours.

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, The Hartford Electric Light Company, Western Massachusetts Electric Company or the 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 State Navy research Laboratory and a desalination pilot operation of the Maxim 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 owners 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 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 Containment

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.