UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 75
License No. DPR-19

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated December 21, 1982, February 1 and 7, and March 24, 1983, as supplemented by letters dated February 24, and March 10, 11, 18, and 31, 1983 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to paragraphs 3.B and 3.M.2, 3, and 4 of Provisional Operating License No. DPR-19 to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 75, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   3.M.2 The Miniumum Critical Power Ratio (MCPR) Safety Limit will be increased 0.03 (TS 1.1.A and 3.3.B.5.C)

   3.M.3 The MCPR Limiting Condition for Operation (LCO) will be increased 0.03 (TS 3.5.K and Fig. 3.5-2)

   3.M.4 The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits will be reduced to 70% of current values for all fuel types.

   (TS reference 3.5.I)

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
1. Changes to the Technical Specifications

Date of Issuance: April 7, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 75

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Make the following page changes to the Appendix A Technical Specifications for License No. DPR-19. The revised pages contain the captioned amendment number and vertical lines indicating the areas of change.

Add Pages	Replace Pages
6A*	1 and 2
11A	5 and 6
15A*	7
22A	10 and 11
36A*	13 - 15
81E-1	16
85B-1	18 - 24 (excluding 18A and 18B; 21a and 21b)
90A	29
	34
	36
	38 and 39
	42
	42A
	46
	47
	58
	61A
	62
	62A
	62B
	63 - 65
	78
	81B
	81B-1
	81C-2 through 81C-5
	81D
	81E
	82
	85A
	85B
	86A
	89
	90
	91e-1
	97

*There are no changes to the provisions contained on these pages; they are included for pagination purposes only.

1.0 DEF TIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. (Deleted)

B. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration.

C. Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of the XN-3 correlation. (Reference XN-NF-512)

D. Hot Standby - Hot standby means operation with the reactor critical, system pressure less than 600 psig, and the main steam isolation valves closed.

E. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

F. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm, or trip. Response time is not part of the routine instrument calibration, but will be checked once per cycle.

G. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument response alarm, and/or initiating action.

H. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

I. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

J. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation the safety limits will never be exceeded.

K. Fraction of Limiting Power Density (FLPD) For fuel fabricated by GE, the fraction of limiting power density is the ratio of the Linear Heat Generation Rate (LHGR) existing at a given location to the design LHGR for that bundle type. FLPD does not apply to ENC fuel.

L. Logic System Function Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

M. Minimum Critical Power Ratio (MCPR) - The minimum in-core critical power ratio corresponding to the most limiting fuel assembly in the core.

N. Mode - The reactor mode is that which is established by the mode-selector-switch.

O. Operable - A system, subsystem, train, component, or device shall be operable when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

P. Operating - Operating means that a system, subsystem, train, component or device is performing its intended functions in its required manner.

Q. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

R. 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 each airlock is closed and sealed.

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

4. All blind flanges and manways are closed.

S. 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 that instrument channel.

1.1 SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

The Safety Limits established to preserve the fuel cladding integrity apply to these variables which monitor the fuel thermal behavior.

Objective

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications

A. Reactor Pressure >800 psig and Core Flow > 10% of Rated.

1 The existence of a minimum critical power ratio (MCPR) less than 1.05 shall constitute violation of the MCPR fuel cladding integrity safety limit.

2.1 LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications

A. Neutron Flux Trip Settings

The limiting safety system trip settings shall be as specified below:

1. APRM Flux Scram Trip Setting (Run Mode)

When the reactor mode switch is in the run position, the APRM flux scram setting shall be:

$$S \leq \left[.58W_D + 62\right]$$

with a maximum set point of 120% for core flow equal to 98 x 10^6 lb/hr and greater, where:

S = setting in per cent of rated power

W_D = per cent of drive flow required to produce a rated core flow of 98 Mlb/hr.

In the event of operation of any fuel assembly fabricated by GE with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

Where: $S \leq (.58W_D + 62) \left[\frac{FRP}{MFLPD}\right]$

FRP = fraction of rated thermal power (2527) MWt)

MFLPD = maximum fraction of limiting power density for GE fuel

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

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING

This adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

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

   When the reactor mode switch is in the refuel startup/hot standby position, the APRM scram shall be set at less than or equal to 15% of rated neutron flux.

1.1 SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psig)

When the reactor pressure is ≤ 800 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25 percent of rated thermal power.

C. Power Transient

1. The neutron flux shall not exceed the scram setting established in Specification 2.1.A for longer than 1.5 seconds as indicated by the process computer.
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.A and a control rod scram does not occur.

D. Reactor Water Level (Shutdown Condition)

Whenever the reactor is in the 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.

*Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

2.1 LIMITING SAFETY SYSTEM SETTING

3. IRM Flux Scram Trip Setting

The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block Setting

The APRM rod block setting shall be:

$$S \leq \left[.58W_D + 50\right]$$

The definitions used above for the APRM scram trip apply.

In the event of operation of any fuel assembly fabricated by GE with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (.58W_D + 50)\left[\frac{FRP}{MFLPD}\right]$$

The definitions used above for the APRM scram trip apply.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0. In which case the actual operating value will be used. The adjustment may also be performed by increasing the APRM gain by the inverse ratio, MFLPD/FRP, which accomplishes the same degree of protection as reducing the trip setting by FRP/MFLPD.

FUEL CLADDING INTEGRITY

The fuel cladding integrity limit is set such that no calculated fuel damages would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the minimum critical power ratio (MCPR) is no less than the MCPR fuel cladding integrity safety limit. MCPR > the MCPR fuel cladding integrity safety limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity by assuring that the fuel does not experience transition boiling.

The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosions or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforation signals a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling, (MCPR of 1.0.) These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity Safety Limit assures that during normal operation and during anticipated operational occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling. See reference XN-NF-524.

A. Reactor Pressure > 800 psig and Core Flow > 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. (Figure 2.1-3).

The MCPR Fuel Cladding Integrity Safety Limit assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at

1.1.A Reactor Pressure > 800 psig and Core Flow > 10% of Rated. (cont'd)

least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR=1.00) and the MCPR Fuel Cladding Integrity Safety Limit is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the XN-3 critical power correlation. Refer to XN-NF-524 for the methodology used in determining the MCPR Fuel Cladding Integrity Safety Limit.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. The assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because boundingly high radial power peaking factors and boundingly flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that during sustained operation at the MCPR Fuel Cladding Integrity Safety Limit there would be no transition boiling in the core. If boiling transition were to occur, however, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U. S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach; much of the data indicates that LWR fuel can survive for an extended period in an environment of transition boiling.

If the reactor pressure should ever exceed the limit of applicability of the XN-3 critical power correlation as defined in XN-NF-512, it would be assumed that the MCPR Fuel Cladding Integrity Safety Limit had been violated. This applicability pressure limit is higher than the pressure safety limit specified in Specification 1.2. For fuel fabricated by General Electric Company, operation is further constrained to a maximum linear heat generation rate (LHGR) of 13.4 kW/ft by Specification 3.5.J. This constraint is established to provide adequate safety margin to 1% plastic strain for abnormal operational transients initiated from high power conditions. Specification 2.1.A.1 provides for equivalent safety margin for transients initiated from lower power conditions by adjusting the APRM flow-biased scram by the ratio of FRP/MFLPD. Specification 3.5.J establishes the maximum value of LHGR which cannot be exceeded during steady power operation for GE fuel types.

For fuel fabricated by Exxon Nuclear Company, (ENC) fuel design criteria have been established to provide protection against fuel centerline melting and cladding strain, ENC has performed

Safety Limit Bases

1.1.A. Reactor Pressure > 800 psig and Core Fuel > 10% of Rated. (cont'd)

fuel design analysis which demonstrate that centerline melting is not predicted to occur during transient overpower conditions throughout the life of the fuel, Protection of the MCPR and MAPLHGR limits and operation within the power distribution assumptions of the fuel design analysis will provide adequate protection against centerline melt and ensures compliance with ENC's clad overstrain criteria for steady state and transient operation. Since ENC's design criteria are more conservative than the 1% plastic strain limitation on GE fuel, the LHGR limitation and APRM scram adjustment for GE fuel established in specifications 3.5.J and 2.1.A.1 respectively are unnecessary for the protection of ENC fuel. The procedural controls of specification 3.1.B will ensure that operation of ENC fuel remains within the power distribution assumptions of the fuel design analysis.

1.1 Safety Limit Bases

1.1.C Power Transient (cont'd)

The computer provided has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C.2 will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel* provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

*Top of active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

2.1 Limiting Safety System Setting Bases

FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the units have been analyzed throughout the spectrum of planned operating conditions up to the rated thermal power condition of 2527 MWt. In addition, 2527 MWt is the licensed maximum steady-state power level of the units. This maximum steady-state power level will never knowingly be exceeded. See reference XN-NF-79-71.

2.1 FUEL CLADDING INTEGRITY (cont'd)

Conservatism is incorporated into the transient analyses which define the MCPR operating limits. Variables which inherently possess little or no uncertainty or whose uncertainty has little or no effect on the outcome of the limiting transient are selected at bounding values. Variables which possess significant uncertainty that may have undesirable effects on thermal margins are addressed statistically. Statistical methods used in the transient analyses are described in XN-NF-81-22. The MCPR operating limits are established such that the occurrence of the limiting transient will not result in the violation of the MCPR Fuel Cladding Integrity Safety Limit in at least 95% of the random statistical combinations of uncertainties. In general, the variables with the greatest statistical significance to the consequences of anticipated operational occurrences are the reactivity feedback associated with the formation and removal of coolant voids and the timing of the control rod scram.

Steady-state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

The bases for individual trip settings are discussed in the following paragraphs.

For analyses of the thermal consequences of the transients, the MCPR's stated in paragraph 3.5.K as the limiting condition of operation bound those which are conservatively assumed to exist prior to initiation of the transients.

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated thermal power. Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

2.1.A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode) cont'd)

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak for G.E. fuel is not increased for any combination of Maximum Fraction of Limiting Power Density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in specification 2.1.A.1. when the MFLPD is greater than the fraction of rated power (FRP).

The adjustment may also be accomplished by increasing the APRM gain by the reciprocal of FRP/MFLPD. This provides the same degree of protection as reducing the trip setting by FRP/MFLPD by raising the initial APRM reading closer to the trip setting such that a scram would be received at the same point in a transient as if the trip setting had been reduced.

2. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the the setpoint and the safety limit, 25 percent of rated. 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. Of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate

2.1.A, Neutron Flux Trip Setting

2. APRM Flux Scram Trip Setting (Refuel or Start and Hot Standby Mode) (cont'd)

to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This switch occurs when reactor pressure is greater than 850 psig.

3. IRM Flux Scram Trip Setting

The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are broken down into 10 ranges, each being one-half of a decade in size.

2.1.A. Neutron Flux Trip Setting

3. IRM Flux Scram Trip Setting (cont'd)

The IRM scram trip setting of 120 divisions is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120 divisions for that range; likewise, if the instrument were on range 5, the scram would be 120 divisions on that range. Thus, as the IRM is ranged up to accomodate the increase in power level, the scram trip setting is also ranged up.

The most significant sources of reactivity change during the power increase are due to control rod withdrawal. In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale.

Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the MCPR fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2.1.B APRM Rod Block Trip Setting

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 gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR fuel cladding integrity safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, 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 100% 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. As with the APRM scram trip setting, the APRM rod block trip setting is adjusted downward or APRM gain increased if the maximum fraction of limiting power density for G.E. fuel exceeds the fraction of rated power, thus preserving the APRM rod block safety margin.

E. Turbine Stop Valve Scram - The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the MCPR fuel cladding integrity safety limit, even during the worst case transient that assumes the turbine bypass is closed.

F. Generator Load Rejection Scram - The generator load rejection 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 the MCPR fuel cladding integrity safety limit for this transient. For the load rejection without bypass transient from 100% power, the peak heat flux (and therefore LHGR) increases on the order of 15% which provides wide margin to the value corresponding to fuel centerline melting and 1% cladding strain.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure - The low pressure isolation at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 850 psig would not necessarily constitute an unsafe condition.

H. Main Steam Line Isolation Valve Closure Scram - The low pressure isolation of the main steam lines at 850 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron flux scram protection over the entire range of applicability of the fuel cladding integrity safety limit. In addition, the isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure, there is no appreciable increase in neutron flux.

1.2 SAFETY LIMIT	2.2 LIMITING SAFETY SYSTEM SETTING

1.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

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

Specification:

The reactor coolant system pressure shall not exceed 1345 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

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

Specification:

A. Reactor Coolant High Pressure Scram shall be $\leq$1060 psig.

B. Primary System Safety Valve Nominal Settings shall be as follows:

1 valve at 1135 psig*
2 valves at 1240 psig
2 valves at 1250 psig
2 valves at 1260 psig
2 valves at 1260 psig

The allowable setpoint error for each valve shall be $\pm$1%.

*Target Rock combination safety/relief valve.

B Lt

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

The pressure safety limit of 1345 psig as measured by the vessel steam space pressure indicator ensures margin to 1375 psig at the lowest elevation of the reactor vessel. The 1375 psig value is derived from the design pressures of the reactor pressel vessel and coolant system piping. The respective design pressures are 1250 psig at 575°F and 1175 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and USASI Code for the reactor coolant system piping. The ASME Boiler and Pressel Vessel Code permits pressure transients up to 10% power over design pressure (110% x 1250 - 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1175 = 1410 psig). The Safety limit pressure of 1375 psig is referenced to the lowest elevation of the reactor vessel. The design pressure for the recirc. suction line piping (1175psig) was chosen relative to the reactor vessel design pressure. Demonstrating compliance of the peak vessel pressure with the ASME overpressure protection limit (1375psig) assures compliance of the suction piping with the USASI limit (1410psig). Evaluation methodology used to assure that this safety limit pressure is not exceeded for any reload is documented in Reference XN-NF-79-71. The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig: this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At that pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safely below the yield strength.

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

The normal operating pressure of the reactor coolant system is 1000psig. For the turbine trip or loss of electrical load transients, the turbine trip scram or generator load rejection scram, together with the turbine bypass system, limit the pressure to approximately 1100psig (2). In addition, pressure relief valves have been provided to reduce the probability of the safety valves, which discharged to the drywell, operating in the event that the turbine bypass should fail.

Finally, the safety valves are sized to keep the reactor vessel peak pressure below 1375 psig with no credit taken for the relief valves during the postulated full closure of all MSIV's without direct (valve position switch) scram. Credit is taken for the neutron flux scram, however,

The indirect flux scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full scale pressure recorder.

(4) SAR, Section II.2.2 -
also: "Dresden 3 Second Reload License Submittal," 9-14-73
also: "Dresden Station Special Report No. 29 Supplement B."

Bases:

2.2 In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the neutron flux scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is available as a backup protection to the direct valve position trip scrams and the high flux scram.

If the high flux scram were to fail, a high pressure scram would occur at 1060 psig. Analyses are performed as described in reference XN-NF-79-71 for each reload to assure that the pressure safety limit is not exceeded.

3.1 LIMITING CONDITION FOR OPERATION	4.1 SURVEILLANCE REQUIREMENT
3.1 REACTOR PROTECTION SYSTEM	**4.1 REACTOR PROTECTION SYSTEM**
Applicability: Applies to the instrumentation and associated devices which initiate a reactor scram.	Applicability: Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.
Objective: To assure the operability of the reactor protection system.	Objective: To specify the type and frequency of surveillance to be applied to the protection instrumentation.
Specification:	Specification:
A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The system response times from the opening of the sensor contact up to and including the opening of the trip actuator contacts shall not exceed 50 milliseconds.	A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.
B. If during operation, the maximum fraction of limiting power density for fuel fabricated by GE exceeds the fraction of rated power when operating above 25% rated thermal power, either:	B. Daily during reactor power operation above 25% rated thermal power, the core power distribution shall be checked for:
a. The APRM scram and rod back settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B. This may be accomplished by increasing APRM gains as described therein.	1. Maximum fraction of limiting power density for fuel fabricated by GE (MFLPD) and compared with the fraction of rated power (FRP).
	2. For compliance with assumptions of the Fuel Design Analysis of overpower conditions for fuel fabricated by ENC.

3.1 LIMITING CONDITIONS FOR OPERATION	4.1 SURVEILLANCE REQUIREMENT

Specifications (cont'd)

b. The power distribution shall be changed such that the maximum fraction of limiting power density no longer exceeds the fraction of rated power.

For fuel fabricated by ENC, operation of the core shall be limited to ensure the power distribution is consistent with that assumed in the Fuel Design Analysis for over-power conditions.

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Minimum Number of Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable			Action*
			Refuel (7)	Startup/Hot Standby	Run	
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
2	High Flux	≤120/125 of Full Scale	X	X	X(5)	A
3	Inoperative		X	X	X(5)	A
	APRM					
2	High Flux	Specification 2.1.A.1	X	X(9)	X	A or B
2	Inoperative		X	X(9)	X	A or B
2	Downscale	≥5/125 of Full Scale	X(12)	X(12)	X(13)	A or B
2	High Flux (15% scram)	Specification 2.1.A.2	X	X	X(14)	A
2	High Reactor Pressure	≤1060 psig	X(11)	X	X	A
2	High Drywell Pressure	≤2 psig	X(8), (10)	X(8), (10)	X(10)	A
2	Reactor Low Water Level	≥1 inch***	X	X	X	A
2	High Water Level in Scram Discharge Tank	≤50 gallons	X(2)	X	X	A
2	Turbine Condenser Low Vacuum	≥23 in. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steamline High Radiation	≤3 X Full Power Background	X(3)	X(3)	X (15)	A or C
4 (6)	Main Steamline Isolation Valve Closure	≤10% Valve Closure	X(3)	X(3)	X	A or C
2	Generator Load Rejection	****	X(4)	X(4)	X(4)	A or C
2	Turbine Stop Valve Closure	≤10% Valve Closure	X(4)	X(4)	X(4)	A or C
2	Turbine Control-Loss of control oil pressure	Greater than or equal to 900 psig	X	X	X	A or C

TABLE 3.1.1 (cont)

Notes:

1. There shall be two operable or tripped trip systems for each function.
2. Permissible to bypass, with control rod block, for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.
3. Permissible to bypass when reactor pressure is < 600 psig.
4. Permissible to bypass when first stage turbine pressure is less than that which corresponds to 45% rated steam flow.
5. IRM's are bypassed when APRM's are onscale and the reactor mode switch is in the run position.
6. The design permits closure of any one valve without a scram being initiated.
7. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:
   a. Mode Switch in Shutdown
   b. Manual Scram
   c. High Flux IRM
   d. Scram Discharge Volume High Level
8. Not required to be operable when primary containment integrity is not required.
9. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).
10. May be bypassed when necessary during purging for containment inerting or deinerting.
11. Not required to be operable when the reactor pressure vessel head is not bolted to the vessel.
12. The APRM downscale trip function is automatically bypassed when the reactor mode switch is in the refuel and startup/hot standby positions.
13. The APRM downscale trip function is automatically bypassed when the IRM instrumentation is operable and not high.

* 14. The APRM 15% scram is bypassed in the run mode.

* 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 insertion of operable rods and complete insertion of all operable rods within four hours.

B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

C. Reduce turbine load and close main steamline isolation valves within 5 hours.

** An APRM will be considered inoperable if there are less than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM's to an APRM.

*** 1 inch on the water level instrumentation is $\geq$ 504" above vessel 0 (See Bases 3.2).

**** Trips upon actuation of the fast closure solenoid which trips the turbine control valves.

15. Due to addition of hydrogen to the primary coolant, the Main Steam Line Radiation monitor setting will be less than or equal to 3 times full power background without hydrogen addition for all conditions except for greater than 20% power with hydrogen being injected during which the Main Steam Line Radiation trip setting will be less than or equal to 3 times full power background with hydrogen addition. Required changes in Main Steam Line Radiation Monitor trip setting will be made within 24 hrs. except during controlled power descensions at which time the setpoint change will be made prior to going below 20% power. If due to a recirculation pump trip or other unanticipated power reduction event the reactor is below 20% power without the setpoint change, control rod motion will be suspended until the necessary trip setpoint adjustment is made.

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 (u-tube in the piping) which accommodates in excess of 50 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 instrument volume which alarm and scram the reactor when the volume of water reaches 50 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 SAR. 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.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity is an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds 3 times full power background for all condition except for greater than 20% power with hydrogen being injected during which the Main Steam Line trip setting is less than or equal to 3 times full power background with hydrogen addition (see note 15). The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors which cause an isolation of the main condensor off-gas line provided the limit specified in Specification 3.8 is exceeded.

The main steamline isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Ref. Section 7.7.1.2 SAR.

a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Nework, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

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

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

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

Experience with passive type instruments in Commonwealth Edison 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 .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.

A comparison of Tables 4.1.1 and 4.1.2 indicates that six instrument channels have not been included in the latter Table. These are: Mode Switch in Shutdown, Manual Scram, High Water Level in Scram Discharge Tank, Main Steam Line Isolation Valve Closure, Generator Load Rejection, and Turbine Stop Valve Closure. All of the devices or sensors associated with these scram functions are simple on-off switches and, hence, calibration is not applicable, i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving, e.g. the switches in the scram discharge volume tank. Based on the above, no calibration is required for these six instrument channels.

B. The MFLPD for fuel fabricated by GE shall be checked once per day to determine if the APRM gains or scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations. 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 MFLPD is adequate.

For fuel fabricated by ENC, the power distribution will be checked once per day to ensure consistency with the power distribution assumptions of the fuel design analysis for overpower conditions. During periods of operation beyond these power distribution assumptions, the APRM gains or scram settings may be adjusted to ensure consistency with the fuel design criteria for overpower conditions.

3.2 LIMITING CONDITION FOR OPERATION	4.2 SURVEILLANCE REQUIREMENT

C. Control Rod Block Actuation

1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3.

2. The minimum number of operable instrument channels specified in Table 3.2.3 for the Rod Block Monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any 30-day period. In addition, one channel may be bypassed above 30% rated power without a time restriction provided that a limiting control rod pattern does not exist and the remaining RBM channel is operable.

D. Steam Jet-Air Ejector Off Gas System

1. Except as specified in 3.2.D.2. below, both steam-jet 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 stack release limit specified in Specification 3.8. The time delay setting for closure of the steam jet-air ejector isolation valves shall not exceed 15 minutes.

3.2 LIMITING CONDITION FOR OPERATION	4.2 SURVEILLANCE REQUIREMENT
2. From and after the date that one of the two steam-jet air ejector off-gas system radiation monitors is made or found to be inoperable, continued reactor power operation is permissible during the next seven days provided the inoperable monitor is tripped in the upscale position.	

TABLE 3.2.1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION FUNCTIONS

Minimum No. of Operable Inst. Channels per Trip System (1)	Instruments	Trip Level Setting	Action (3)
2	Reactor Low Water	>144" above top of active fuel *	A
2	Reactor Low Low Water	≥ 84" above top of active fuel *	A
2	High drywell pressure	≤2 psig rated (4), (5)	A
2 (2)	High Flow Main Steam line	≤120% of rated steam flow	B
2 of 4 in each of 4 sets	High Temperature Main Steam Line Tunnel	≤200°F	B
2	High Radiation Main Steam Line Tunnel (6)	≤ 3 times full power background (see note 7)	B
2	Low Pressure Main Steamline	≥850 psig	B
	High Flow Isolation Condenser Line		
1	Steamline Side	≤20 psi diff. on steamline side	C
1	Condensate Return Side	≤32" water diff. on condensate return side	C
2	High Flow HPCI Steam Line	≤150" water	D
4	High Temperature HPCI Steam Line Area	≤200°F	D

Notes:

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.

*Top of active fuel is defined as 360" above vessel zero for all water levels used in the LOCA Analysis (See Bases 3.2).

TABLE 3.2.1 (cont)

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.

D. Close isolation valves in HPCI subsystem.

4. Need not be operable when primary containment integrity is not required.

5. May be bypassed when necessary during purging for containment inerting and deinerting.

28/19

6. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in the primary coolant.

7. Due to addition of hydrogen to the primary coolant, the Main Steam Line Radiation monitor setting will be less than or equal to 3 times full power background without hydrogen addition for all conditions except for greater than 20% power with hydrogen being injected during which the Main Steam Line Radiation trip setting will be less than or equal to 3 times full power background with hydrogen addition. Required changes in Main Steam Line Radiation Monitor trip setting will be made within 24 hrs. except during controlled power descensions at which time the setpoint change will be made prior to going below 20% power. If due to a recirculation pump trip or other unanticipated power reduction event the reactor is below 20% power without the setpoint change, control rod motion will be suspended until the necessary trip setpoint adjustment is made.

DPR-19

INSTRUMENTATION THAT INITIATES ROD BLOCK

Table 3.2.3

Minimum No. of Operable Inst. Channels Per Trip System(1)	Instrument	Trip Level Setting	
1	APRM upscale (flow bias)(7)	$\leq \left[0.58W_D + 50\right]\frac{FRP}{MFLPD}$	(2)
1	APRM upscale (refuel and Startup/Hot Standby mode)	≤12/125 full scale	
2	APRM downscale (7)	≥3/125 full scale	
1	Rod block monitor upscale (flow bias)(7)	$\leq$ 0.65 W_D + 45	(2)
1	Rod block monitor downscale (7)	≥5/125 full scale	
3	IRM downscale (3)	≥5/125 full scale	
3	IRM upscale	≤108/125 full scale	
3	IRM detector not fully inserted in the core		
2(5)	SRM detector not in startup position	(4)	
2(5)(6)	SRM upscale	$\leq 10^5$ counts/sec	
1	Scram Discharge Volume Water Level - High	25 gal.	

NOTES:

1. For the Startup/Hot Standby and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function, except the SRM rod blocks, IRM upscale, IRM downscale and IRM detector not fully inserted in the core need not be operable in the "Run" position and APRM downscale, APRM upscale (flow bias), and RBM downscale need not be operable in the Startup/Hot Standby mode. The RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, if the first column cannot be met for both trip systems, the systems shall be tripped. For the Scram Discharge Volume water level high rod block, there is one instrument channel for one trip system.

2. W_D percent of drive flow required to produce a rated core flow of 98 Mlb/m.
MFLPD=highest value of FLPD for G.E. fuel.

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

4. This function may be bypassed when the count rate is $\geq$100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

7. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

Bases:

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

Some of the settings on the instrumentation that initiates or controls core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

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

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

The low-reactor water level instrumentation is set to trip at >8 inches on the level instrument (top of active fuel is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above the top of active fuel. Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the LOCA analyses.
This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR Section 7.7.2). For a trip setting of 504 inches above vessel zero (144 inches above top of active fuel) and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break; the setting is therefore adequate.

The low low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, -59 inches is 84 inches above the top of active fuel). This trip initiates closure of Group 1 primary containment isolation valves, Ref. Section 7.7.2.2 SAR, and also activates the ECC subsystems, starts the emergency diesel generator and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria.

The instrumentation also covers the full range or spectrum of breaks and meets the above criteria.

This 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. 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 start-up. 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 Group 1 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 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 outside the drywell, thus only 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 valve closure, limit the mass inventory loss such that fuel is not uncovered, fuel temperatures remain less than 1500°F and release of radioactivity of to the environs is well below 10 CFR 100 guidelines. Ref. Sections 14.2.3.9 and 14.2.3.10 SAR.

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 a 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 for this accident. With the established setting of 3 times full power background for all conditions except for greater than 20% power with hydrogen being injected during which the Main Steamline trip setting is less than or equal to 3 times full power background with hydrogen addition, and main steamline isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Ref. Section 14.2.1.7 SAR. The performance of the process radiation monitoring system relative to detecting fuel leakage shall be evaluated during the first five years of operation. The conclusions of this evaluation will be reported to the Atomic Energy Commission.

Pressure instrumentation is provided which trips when main steamline pressure drops below 850 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "Refuel" and "Startup/Hot Standby" mode this trip function is bypassed. This function is provided to pro-

3.3 LIMITING CONDITION FOR OPERATION

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	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

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	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

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

4.3 SURVEILLANCE REQUIREMENTS

C. Scram Insertion Times

1. After each refueling outage and prior to power operation with reactor pressure above 800 psig, all control rods shall be subject to scram-time tests from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

2. At 16 week intervals, 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% of the control rod drives have been scram tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. Following completion of each set of scram testing as described above, the results will be compared against the average scram speed distribution used in the transient analysis to verify the applicability of the current MCPR Operating Limit. Refer to Specification 3.5.K.

indicative of a generic control rod drive problem and the reactor will be shutdown. Also, if damage within the control rod drive mechanism and, in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWR's. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

3. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume. This ensures that water accumulation does not occur which would cause an early termination of control rod movement during a full core scram. These specifications provide for the periodic verification that the valves are open and for testing of these valves under reactor scram conditions during each Refueling Outage.

B. **Control Rod Withdrawal**

1. Control rod dropout accidents as discussed in Reference XN-NF-80-19, Vol. I, can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would provide cause for suspecting a rod to be uncoupled and stuck. Restricting recoupling verifications to power levels above 20% provides assurance that a rod drop during a recoupling verification would not result in a rod drop accident.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted, and if an adequate shutdown margin with one control rod withdrawn has been demonstrated since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod sequences which are withdrawn could not be worth enough to cause the rod drop accident design limit of 280 cal/gm to be exceeded if they were to drop out of the core in the manner defined for the Rod Drop Accident. These sequences are developed prior to initial operation

of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM or a second qualified station employee. These sequences are developed to limit reactivity worths of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content, 425 cal/gm, at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 1.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2,1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

Bases(cont'd)

Parametric Control Rod Drop Accident analyses have shown that for wide ranges of key reactor parameters (which envelope the operating ranges of these variables), the fuel enthalpy rise during a postulated control rod drop accident remains considerably lower than the 280 cal/gm limit. For each operating cycle, cycle-specific parameters such as maximum control rod worth, Doppler coefficient effective delayed neutron fraction and maximum four-bundle local peaking factor are compared with the results of the parametric analyses to determine the peak fuel rod enthalpy rise. This value is then compared against the Technical Specification limit of 280 cal/gm to demonstrate compliance for each operating cycle. If cycle specific values of the above parameters are outside the range assumed in the parametric analyses, an extension of the analysis or a cycle specific analysis may be required. Conservatism present in the analysis, results of the parametric studies, and a detailed description of the methodology for performing the Control Rod Drop Accident analysis are provided in reference XN-NF-80-19, Volume 1 (Supplements 1 and 2).

Bases (con'd)

The Rod Worth Minimizer provides automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Ref. Section 7.9 SAR. It serves as a backup to procedural control o control rod worth. In the event that the Rod Worth Minimizer is out of service, when required, a licensed operator or other qualified technical employee can manually fulfill the control rod pattern conformance functions of the Rod Worth Minimizer. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperbbility recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 20% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is requried to be operating during a startup for the withdrawal of a significant number control rods for any startup after June 1, 1974.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the

operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws rods according to a written sequence. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists. Amendments 17/18 and 19/20 present the results of an evaluation of a rod block monitor failure. These amendments show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPRs less than the MCPR fuel cladding integrity safety limit. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns.

C. Scram Insertion Times

The performance of the control rod insertion system is analyzed to verify the system's ability to bring the reactor subcritical at a rate fast enough to prevent violation of the MCPR Fuel Cladding Integrity Safety Limit and thereby avoid fuel damage. The analyses demonstrate that if the reactor is operated within the limitations set in Specification 3.5.K, the negative reactivity insertion rates associated with the observed scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR safety limit.

In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 millisecons. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram value solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses, and is also included in the allowable scram insertion times specified in Specification 3.3.C. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Fifty percent of the control rod drives in the reactor are tested every sixteen weeks to verify adequate performance. Observed plant data were used to determine the average scram performance used in the transient anslyses.

Scram Insertion Times (cont'd)

and the results of each set of control rod scram tests during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly. If an individual test or group of tests should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a redetermination of thermal margin requirements is undertaken (as required by Specification 3.5.K) unless it can be shown that the number of individual drives failing outside the statistical population defining the nominal performance is less than the allowable number of inoperable control rod drives. If the number of statistically aberrant drives falls within this limitation, operation will be allowed to continue without redetermination of thermal margin requirements provided the identified aberrant drives are fully inserted into the core and deenergized in the manner of an inoperable rod drive.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no reassessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 second is greater than 0.999 for a normal distribution.

D. Control Rod Accumulators

The basis for this specification was not described in the SAR and, therefore, is presented in its entirety. Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a $k_{eff} < 1.0$ -- other repeating rod sequences with more rods withdrawn resulted in $k_{eff} > 1.0$. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

E. Reactivity Anomalies

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

G. Economic Generation Control System

Operation of the facility with the Economic Generation Control System with automatic flow control is limited to the range of 65-100% of rated core flow. In this flow range and with reactor power above 20% the reactor can safely tolerate a rate of change of load of 8 MW(e)/sec. (Reference FSAR Amendment 9 - Unit 2, 10-Unit 3). Limits within the Economic Generation Control System and Reactor Flow Control System preclude rates of change greater than approximately 4 MWe/sec.

When the Economic Generation Control System is in operation, this fact will be indicated on the main control room console. The results of initial testing will be provided to the AEC at the onset of routine operation with the Economic Generation Control System.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT
D. Automatic Pressure Relief Subsystems	D. Surveillance of the Automatic Pressure Relief Subsystem shall be performed as follows:
1. Except as specified in 3.5.D.2 and 3 below, the Automatic Pressure Relief Subsystem shall be operable whenever the reactor pressure is greater than 90 psig and irradiated fuel is in the reactor vessel.	1. During each operating cycle the following shall be performed: a. A simulated automatic initiation which opens all pilot valves, and b. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure. c. A logic system functional test shall be performed each refueling outage.
2. From and after the date that one of the five relief 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 HPCI Subsystem is operable.	2. When it is determined that one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI shall be demonstrated to be operable immediately and weekly thereafter
3. From and after the date that more than one of five relief valves of the automatic pressure relief subsystem 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 24 hours unless repairs are made and provided that during such time the HPCI Subsystem is operable.	3. When it is determined that more than one relief valve of the automatic pressure relief subsystem is inoperable, the HPCI subsystem shall be demonstrated to be operable immediately

3.5 LIMITING CONDITION FOR OPERATION

I. Average Planar LHGR

During steady state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure for G.E. fuel and average bundle exposure for Exxon fuel at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5-1. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR 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.

4.5 SURVEILLANCE REQUIREMENT

I. Average Planar Linear Heat Generation Rate (APLHGR)

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

3.5 LIMITING CONDITION FOR OPERATION

3.5.J LOCAL LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly fabricated by GE at any axial location shall not exceed the design value of 13.4 kw/ft.

If at any time during operation, it is determined by normal surveillance that the limiting value for LHGR for G.E. fuel is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR 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

4.5 SURVEILLANCE REQUIREMENT

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at ≥ 25% rated thermal power.

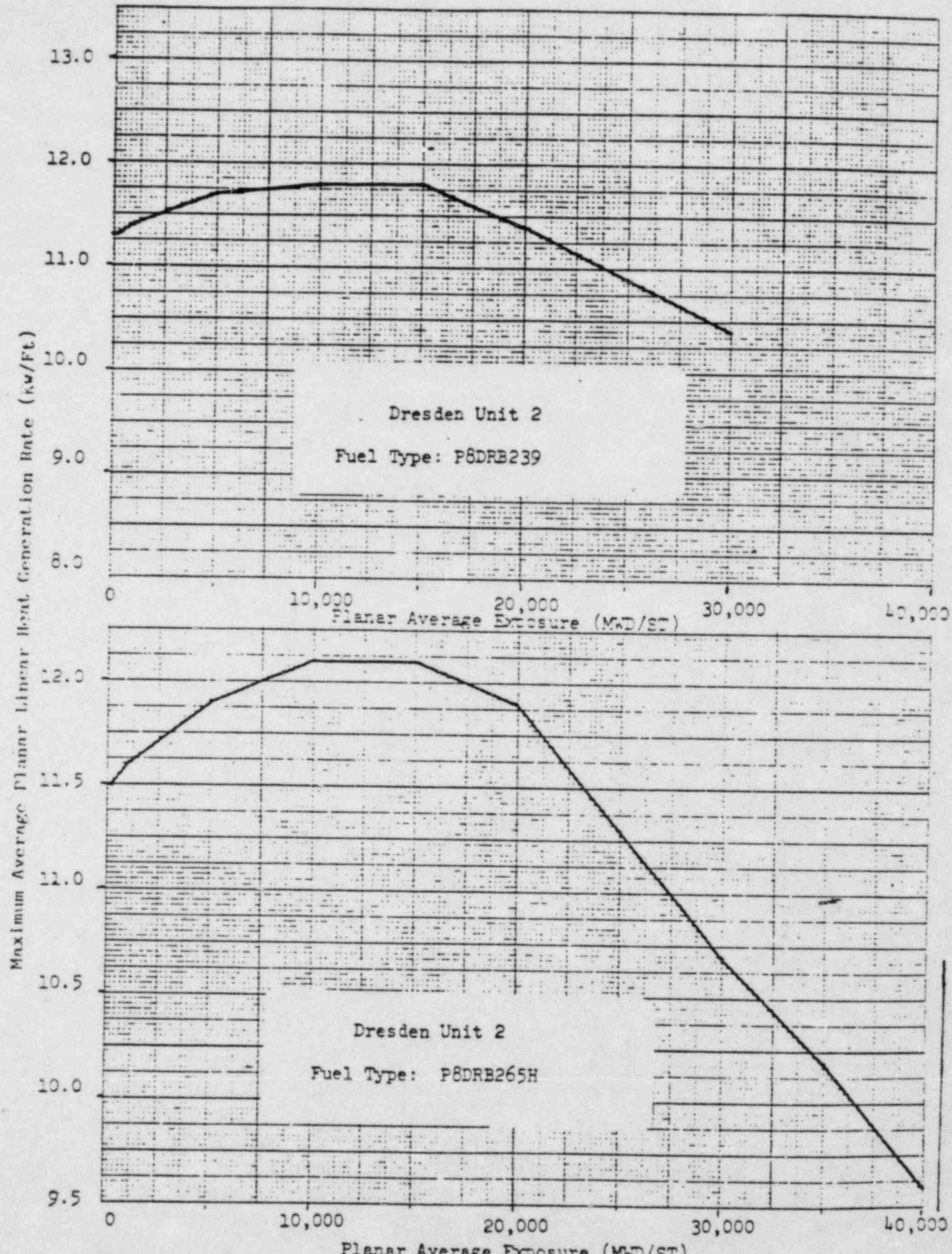
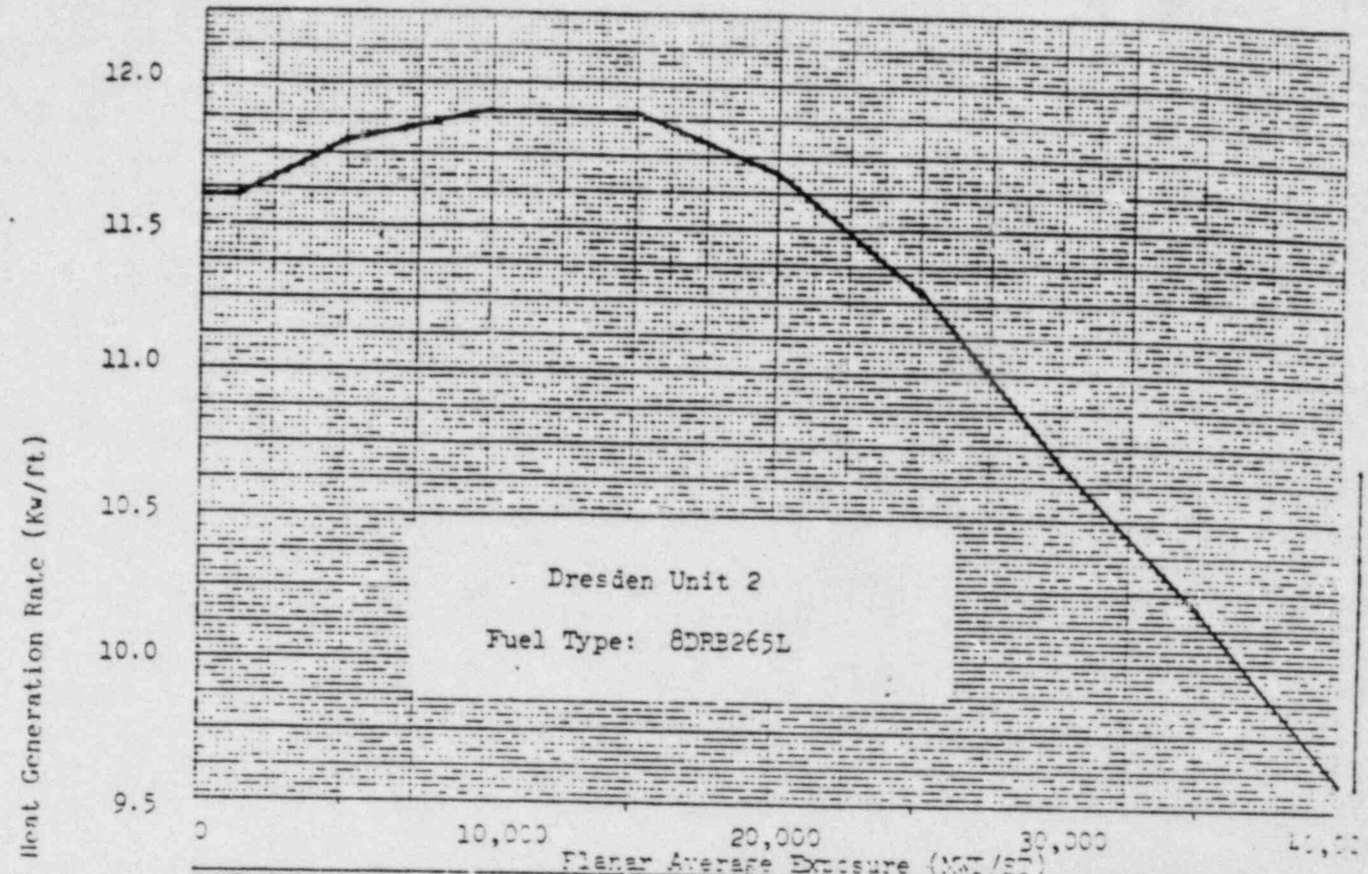


Figure 3.5-1
(Sheet 2 of 5)

Maximum Average Planar Linear
Heat Generation Rate (MAPLHGR)
vs. Planar Average Exposure

81 C-2



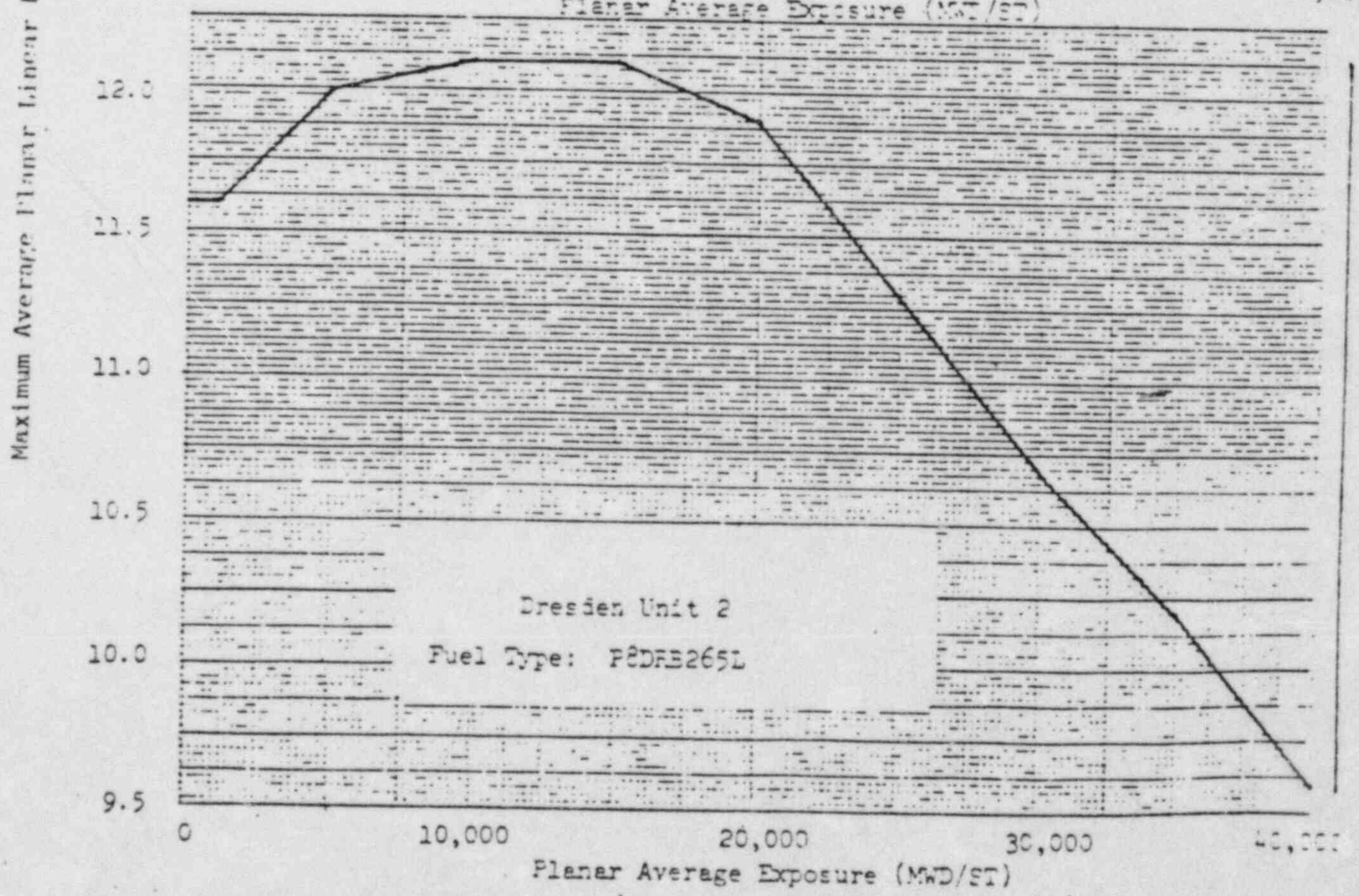
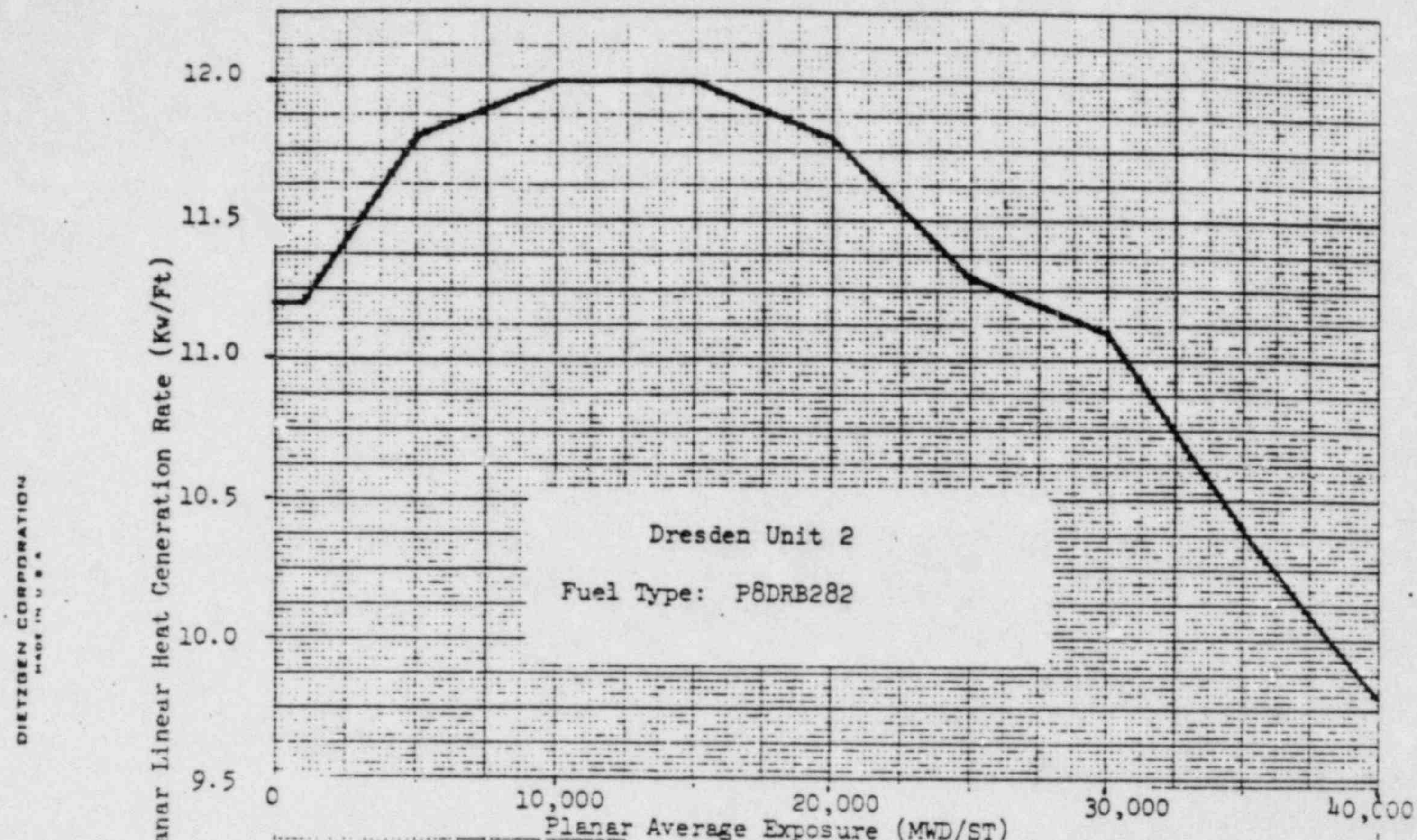


Figure 3.5-1
(Sheet 3 of 5)

Maximum Average Planar Linear
Heat Generation Rate (MAPLHGR)
vs. Planar Average Exposure



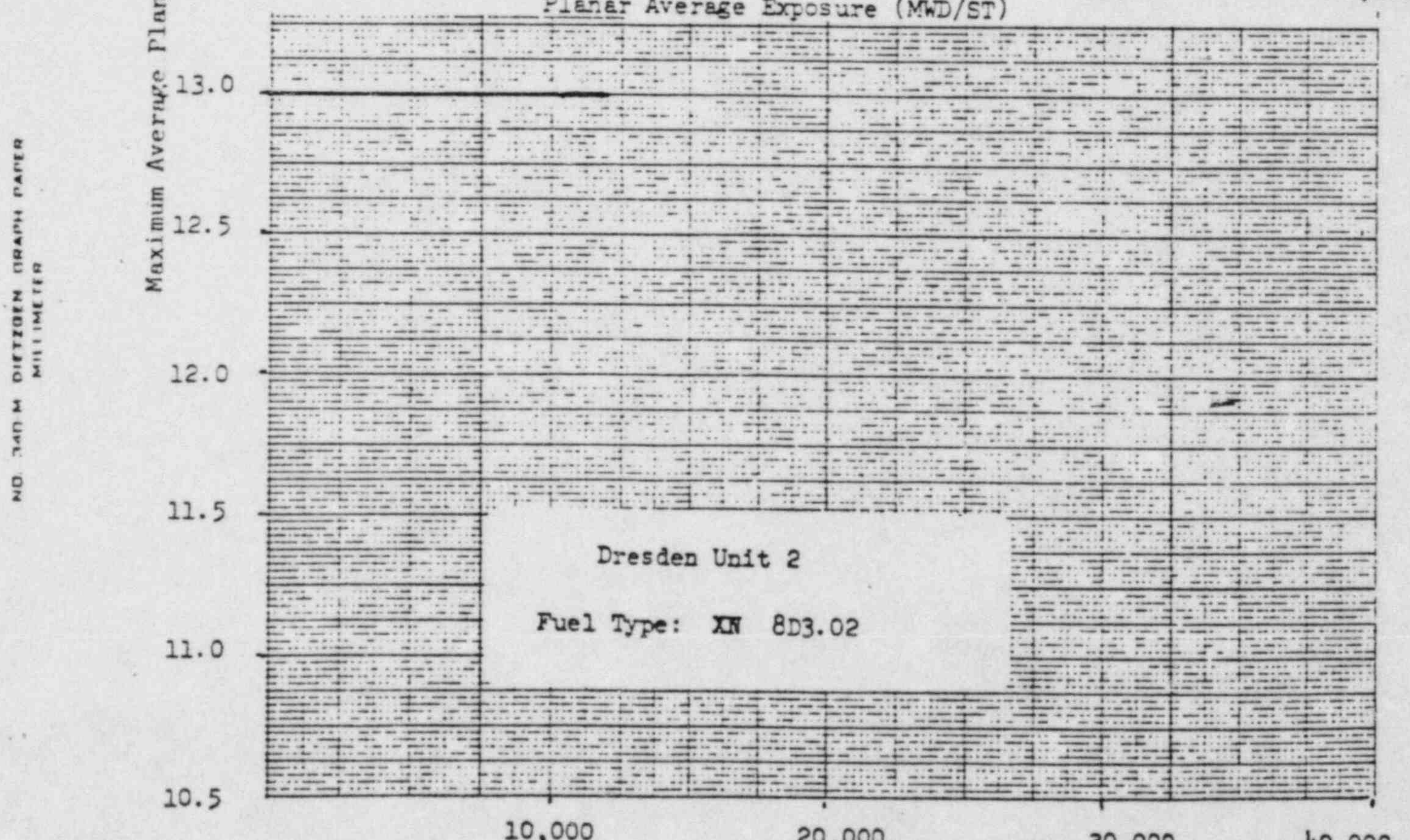


Figure 3.5-1
(Sheet 4 of 5)

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)
Versus Exposure

81 C-4

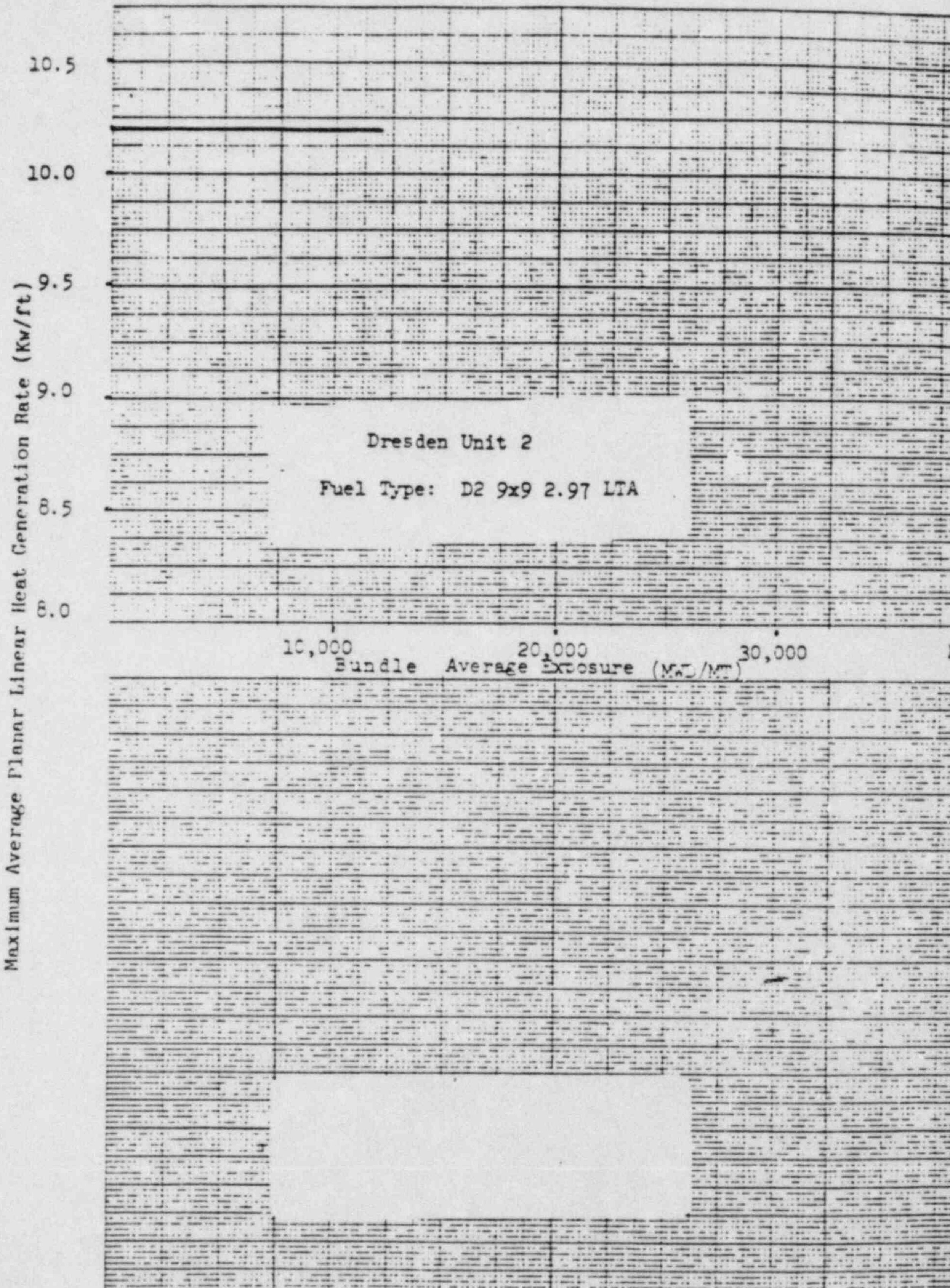


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Figure 3.5-1
(Sheet 5 of 5)

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Bundle Average Exposure

3.5 LIMITING CONDITIONS FOR OPERATION	4.5 SURVEILLANCE REQUIREMENTS

K. Minimum Critical Power Ratio (MCPR)

During steady state operation at rated core flow, MCPR shall be greater than or equal to -

1.34 for XN-1 8x8 and G.E. 8x8 Fuel types

1.35 for G.E. 8x8R
1.38 for XN-1 9x9 LTA

For core flows other than rated, the MCPR operating limit shall be as follows:

1. Manual Flow Control-the MCPR Operating Limit shall be the value from Figure 3.5-2 sheet 1 or the above rated core flow value, which ever is greater.
2. Automatic Flow Control-the MCPR Operating Limit shall be the value from Figure 3.5-2 Sheet 1, Sheet 2 or the above rated core flow value, whichever is greatest.

If at any time during steady state power operation, it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR 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.

In the event the average 90% scram insertion time determined by Spec 3.3.C for all operable control rods exceeds 2.74 seconds, the MCPR limit shall be increased by the amount equal to (0.092T-0.252) where T equals the average 90% scram insertion time for the most recent half-core or full-core surveillance data from Spec. 4.3.C.

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at ≥25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

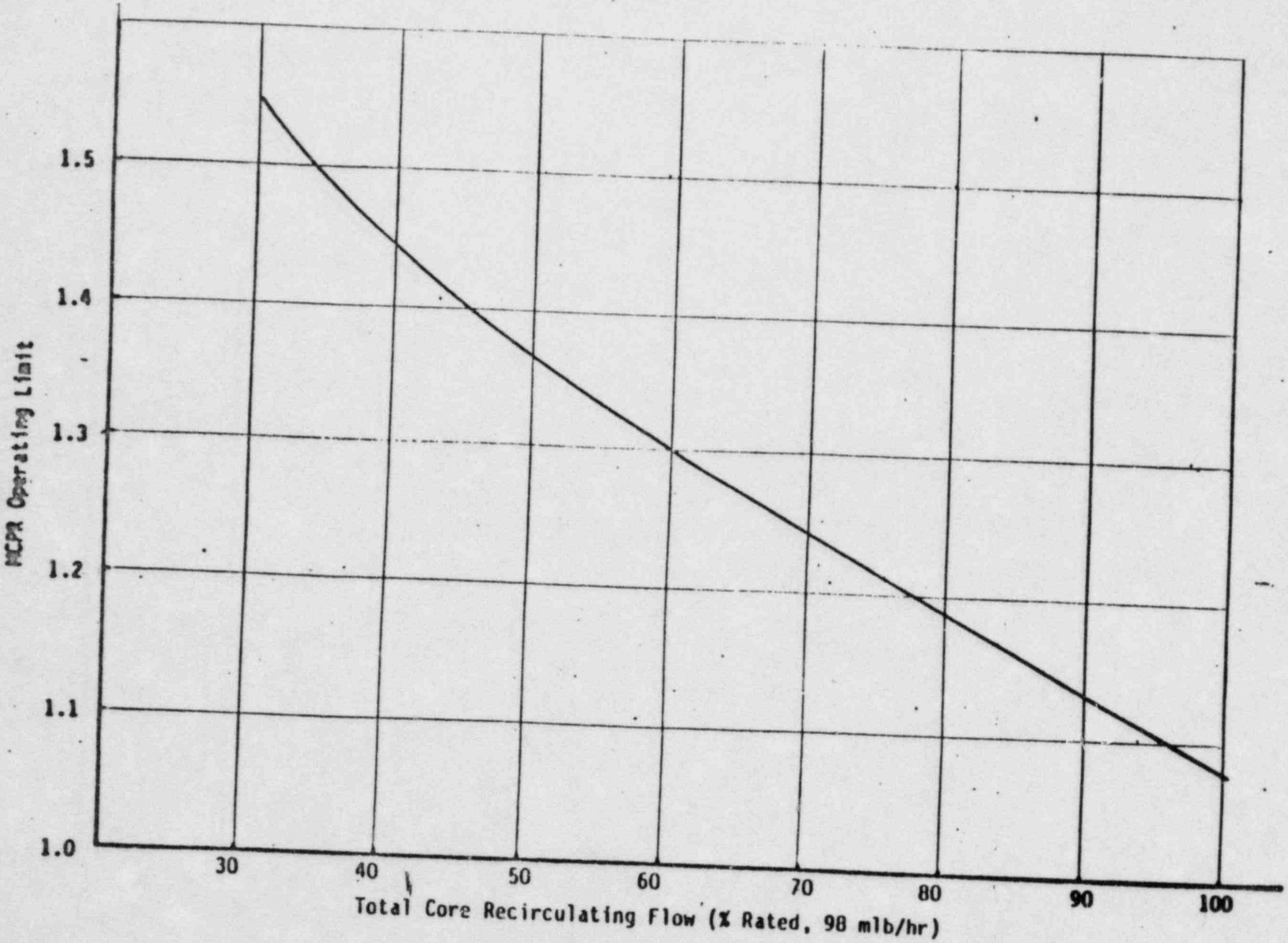

Figure 3.5-2 (Sheet 1 of 2) MCPR Limit for Reduced Core Flow

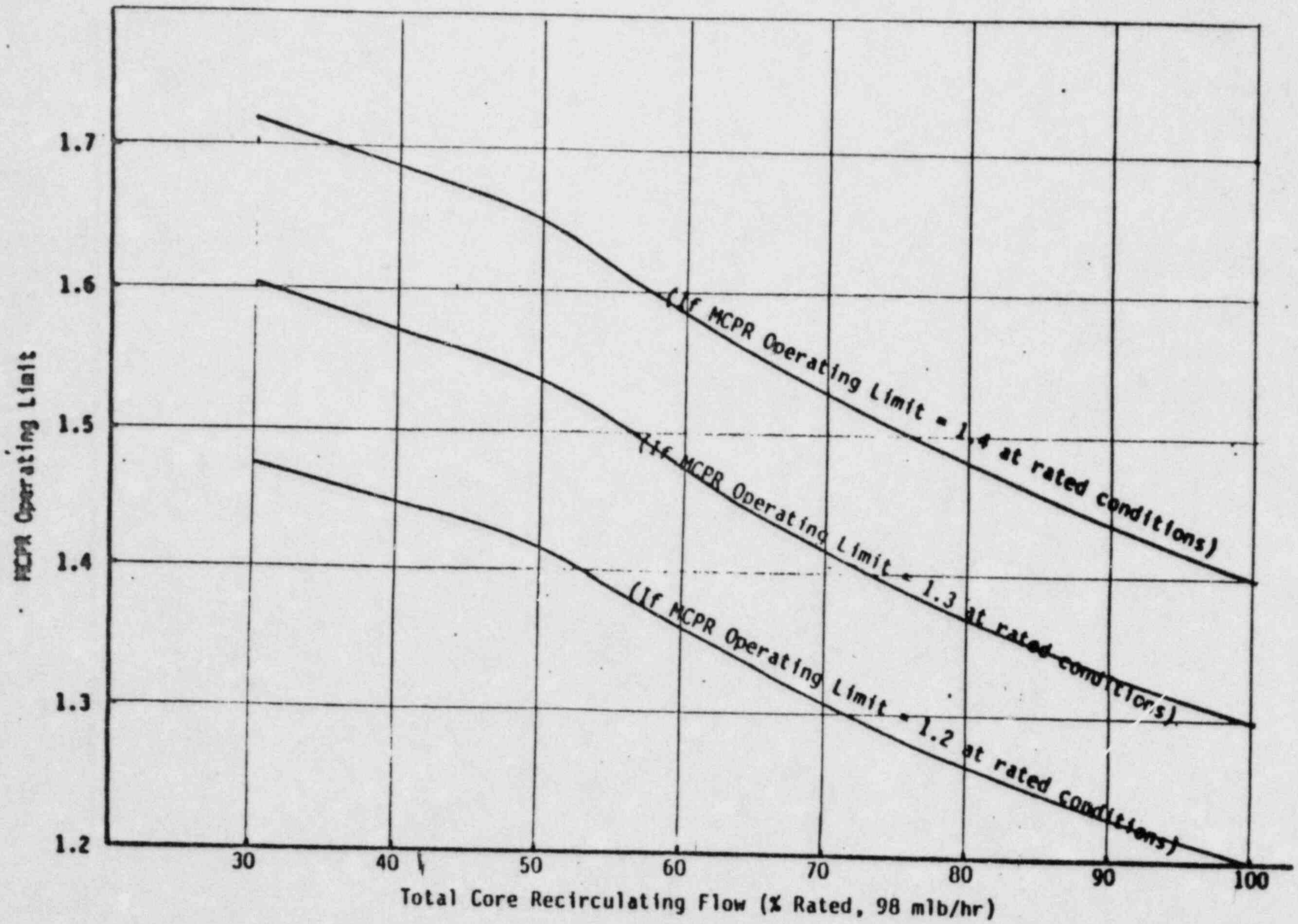

Figure 3.5-2 (Sheet 2 of 2) MCPR Limit for Automatic Flow Control

3.5 Limiting Conditions for Operation Bases

A. Core Spray and LPCI Mode of the RHR System - This specification assures that adequate emergency cooling capability is available.

Based on the loss of coolant analyses included in References (1), (2) and (4) in accordance with 10CFR50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit the calculated peak clad temperature to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

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 (3). 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 and to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in reference 3 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.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the

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(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision1, April 1979.

(2) NEDO-20566, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K.

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

(4) XN-NF-82-88 "Dresden Unit 2 LOCA Analysis Using the ENC EXEM/BWR Evaluation Model MAPLHGR Results"

3.5 Limiting Condition for Operation Bases (Cont'd)

I. Average Planar LHGR

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10CFR50 Appendix K considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average LHGR of all the rods in a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within a fuel 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 planar LHGR is sufficient to assure that calculated temperatures are below the 10CFR50, Appendix K limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in Reference (1) and in reference XN-NF-82-88. Power operation with APLHGRs at or below those shown in Fig. 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit.

The maximum average planar LHGRs for G.E. fuel plotted in Fig. 3.5.1 at higher exposures result in a calculated peak clad temperature of less than 2200°F. However the maximum average planar LHGRs are shown on Fig. 3.5.1 as limits because conformance calculations have not been performed to justify operation at LHGRs in excess of those shown.

J. Local LHGR

This specification assures that the maximum linear heat generation rate in any fuel rod fabricated by G.E. is less than the design linear

(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision 1, April, 1979.

Amendment No. 58, 75

heat generation rate even if fuel pellet densification is postulated.

For fuel fabricated by ENC, protection of the MCPR and MAPLHGR limits and operation within the power distribution assumptions of the Fuel Design Analysis provides adequate protection against cladding strain limits, hence the LHGR limitation for GE fuel is unnecessary for the protection of ENC fuel.

The steady-state values for MCPR specified in the Specification were determined using the THERMEX thermal limits methodology described in XN-NF-80-19, Volume 3. The safety limit implicit in the Operating limits is established so that during sustained operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The Limiting Transient Δ CPR implicit in the operating limits was calculated such that the occurrence of the limiting transient from the operating limit will not result in violation of the MCPR safety limit in at least 95% of the random statistical combinations of uncertainties.

Transient events of each type anticipated during operation of a BWR/3 were evaluated to determine which is most restrictive in terms of thermal margin requirements. The generator load rejection/turbine trip without bypass is typically the limiting event. The thermal margin effects of the event are evaluated with the THERMEX Methodology and appropriate MCPR limits consistent with the XN-3 critical power correlation are determined. Several factors influence which transient results in the largest reduction in critical power ratio, such as the cycle-specific fuel loading, exposure and fuel type. The current cycle's reload licensing analyses identifies the limiting transient for that cycle.

As described in specification 4.3.C.3 and the associated Bases, observed plant data were used to determine the average scram performance used in the transient analyses for determining the MCPR Operating Limit. If the current cycle scram time performance falls outside of the distribution assumed in the analyses, an adjustment of the MCPR limit may be required to maintain margin to the MCPR Safety Limit during transients. Compliance with the assumed distribution and adjustment of the MCPR Operating Limit will be performed in accordance with Technical Specifications 4.3.C.3 and 3.5.K.

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the MCPR Safety Limit in the event of an uncontrolled recirculation flow increase to the physical limit of pump flow. This protection is provided for manual and automatic flow control by choosing the MCPR operating limit as the value from Figure 3.5-2 Sheet 1 or the rated core flow value, whichever is greater. For Automatic Flow Control, in addition to protecting the MCPR Safety Limit during the flow run-up event, protection is provided against violating the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow. This protection is provided by the reduced flow MCPR limits shown in Figure 3.5-2 Sheet 2 where the curve corresponding to the current rated flow MCPR limit is used (linear interpolation between the MCPR limit lines depicted is permissible).

Therefore, for Automatic Flow Control, the MCPR Operating Limit is chosen as the value from Figure 3.5-2 Sheet 1, Sheet 2 or the rated flow value, whichever is greatest. It should be noted that if the rated flow MCPR Limit must be increased due to degradation of control rod scram times during the current cycle, the new value of the rated flow MCPR limit is applied when using Figure 3.5-2 Sheet 2.

4.5 Surveillance Requirements Bases (cont'd)

I. Average Planar LHGR

At core thermal power levels less than or equal to 25 per cent, operating plant experience and thermal hydraulic analyses indicate that the resulting average planar LHGR is below the maximum average planar LHGR by a considerable margin; therefore, evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25 per cent rated thermal power is sufficient since power distribution shifts are slow when there have not been significant power or control rod changes.

J. Local LHGR

The LHGR for G.E. fuel shall be checked daily during reactor operation at greater than or equal to 25 per cent power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by a considerable margin when employing a per missible control rod pattern below 25% rated thermal power.

K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 per cent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicates that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

In addition, the K_f correction applied to the LCO provides margin for flow increase from low flows.

3.6 LIMITING CONDITION FOR OPERATION	4.6 SURVEILLANCE REQUIREMENT

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in 3.6.C.3:

Conductivity	2 μmho/cm
Chloride ion	0.1 ppm

3. For reactor startups the maximum value for conductivity shall not exceed 10 μmho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm, for the first 24 hours after placing the reactor in the power operating condition.

4. Except as specified in 3.6.C.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hour.

Conductivity	5 μmho/cm
Chloride ion	0.5 ppm

5. If Specification 3.6.C.1, 3.6.C.2, 3.6.C.3 or 3.6.C.4 is not met, an orderly shutdown shall be initiated.

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212^0F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

2. During startups and at steaming rates below 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.

3. a. With steaming rates greater than or equal to 100,000 pounds per hour, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes) and analyzed for conductivity and chloride ion content.

   b. When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least daily and analyzed for conductivity and chloride ion content.

D. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system. Sump flow monitoring and recording shall be performed once per 4 hours. Air sampling shall be performed once per day.

3.4 LIMITING CONDITION FOR OPERATION

2. After completion of the investigation or containment inspection, specified in 4.6.D.2.a or 4.6.D.2.b, if the leakage is determined to be due to a thru wall pipe crack on the reactor coolant pressure boundary, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

E. **Safety and Relief Valves**

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320°F, all nine of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be less than or equal to 90 psig and less than or equal to 320°F within 24 hours.

4.6 SURVEILLANCE REQUIREMENT

2. The following additional leakage limits shall be met until the recirculation piping indications have been resolved.

Whenever the reactor is at operating pressure, the following will apply to unidentified leakage;

a. If a 1 gpm increase over the previous 4 hours occurs or when leakage equals 3 gpm total, an investigation of the cause of the leakage increase will be performed. This investigation should consist of taking drywell air and water samples, and a review of any previous plant evolutions to the extent necessary to determine the source of leakage.

b. If leakage equals 4 gpm, a containment inspection will be conducted to determine the source of leakage.

E. **Safety and Relief Valves**

A minimum of ½ of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (psig)
1	1135*
2	1240
2	1250
2	1260
2	1260

The allowable set point error for each valve is $\pm 1\%$.

3.4 LIMITING CONDITION FOR OPERATIONS

4.6 SURVEILLANCE REQUIREMENT

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Valve No.	Set Point (psig)
203-3A	1124*
203-3B	1101
203-3C	1101
203-3D	1124
203-3E	1124

* Target Rock combination safety/relief valve.

The allowable set point error for each valve is ± 1%.

TABLE 3.6.1b

SAFETY RELATED MECHANICAL SNUBBERS

SNUBBER NO.	LOCATION	ELEVATION	AZIMUTH	SNUBBER IN HIGH RADIATION AREA DURING SHUTDOWN	SNUBBERS INACCESSIBLE DURING NORMAL OPERATION	SNUBBERS ACCESSIBLE DURING NORMAL OPERATION
1	Drywell Recirc. Motor 2B-202	524'	328°	X	X	
2	Drywell Recirc. Motor 2B-202	524'	302°	X	X	
3	Drywell Recirc. Motor 2B-202	524'	315°	X	X	
4	Drywell Recirc. Motor 2A-202	524'	148°	X	X	
5	Drywell Recirc. Motor 2A-202	524'	122°	X	X	
6	Drywell Recirc. Motor 2A-202	524'	135°	X	X	
7	Drywell Recirc. Pump 2B-202	512'	326°	X	X	
8	Drywell Recirc. Pump 2B-202	512'	304°	X	X	
9	Drywell Recirc. Pump 2B-202	517'	315°	X	X	
10	Drywell Recirc. Pump 2A-202	512'	124°	X	X	
11	Drywell Recirc. Pump 2A-202	512'	146°	X	X	
12	Drywell Recirc. Pump 2A-202	507'	135°	X	X	
13-16	Removed					
17	Drywell Recirc Header 201B-22"	533'6"	195°	X	X	
18-20	Removed					
21	Drywell Recirc Header 201A-22"	533'6"	22°	X	X	
22-23	Removed					
24	Drywell Feedwater Line 3204D-12"	538'	108°	X	X	
25-29	Removed					
30	Drywell Core Spray Line 1403-10"	575'	336°	X	X	
31	Drywell Core Spray Line 1404-10"	562'	231°	X	X	
32	Drywell Target Rock Valve 203-3A	542'6"	16°	X	X	
33	Drywell Target Rock Valve 203-3A	542'4"	31°	X	X	
34	Drywell Target Rock Valve 203-3A	540'0"	19°	X	X	
35	Removed					
36	Drywell Recirc. Line 201B-28"	518'	270°	X	X	

Modifications to this table due to changes in high radiation should be submitted to the NRC as part of the next license amendment request.

the operator for correcting the off-standard condition include, operation of the reactor clean-up system, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the clean-up system to re-establish the purity of the reactor coolant. During start-up periods, which are in the category of less than 100,000 pounds per hours, conductivity may exceed 2 μmho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 μmho (other than short term spikes), samples shall be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity at the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.C.3 may be performed by a gamma scan.

D. Coolant Leakage - Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite a-c power. The normally expected back-ground leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.D on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm as specified in 3.6.D, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shutdown to allow further investigation and corrective action.

The additional leakage requirements will be in effect only while the reactor is operated with the recirculation flaws detected during the 1983 Refueling Outage. The additional leakage requirements will provide more conservative detection and corrective action should the current flaws propagate thru wall.

The capacity of the drywell sump is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.