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Am. 34 1/30/78 2. E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Dresden Nuclear Power Station, Units Nos. 1, 2 and 3.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations; 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50-54 and 50-59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

Commonwealth Edison is authorized to operate the facility at steady state power levels not in excess of 2527 megawatts (thermal).

B. Technical Specifications

Am. ~~98~~ TBD ~~2/19/88~~

The Technical Specifications contained in Appendix A, as revised through Amendment ~~98~~ TBD are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. Reports

Commonwealth Edison shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

Commonwealth Edison shall keep facility operating records in accordance with the requirements of the Technical Specifications.

E. For the purpose of repairing a crack in the recirculation bypass line in the "A" loop, the licensee may perform the repair program as described in a report entitled "Commonwealth Edison Company Dresden Station 2A Recirculations Pump 4 Equalizing Line Repair Program" transmitted by letter dated September 23, 1974.

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3. F. Restrictions

Operation in the coastdown mode is permitted to 40% power.
~~power. Should off-normal feedwater heating be necessary for extended periods during coastdown (i.e. greater than 24 hours) the Licensee shall perform a safety evaluation to determine if the MCPR Operating Limit and calculated peak pressure for the worst case abnormal operating transient remain bounding for the new condition.~~

Am. 58
3/31/81

G. Equalizer Valve Restriction

Am. 21
5/23/77

The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation.

H. The licensee may proceed with and is required to complete the modification identified in Paragraphs 3.1.1 through 3.1.23 of the NRC's Fire Prtotection Safety Evaluation (SE) dated March 1978 on the facility. All modifications are to be completed by start-up following the 1979 Unit 2 refueling outage. In addition, the licensee shall submit the additional information identified in Table 3.1 of this SE in accordance with the schedule contained therein. In the event these dates for submittal cannot be met, the licensee shall submit a report, explaining the circumstances, together with a revised schedule.

Am. 36
3/22/78

I. Physical Protection

The licensee shall full implement and maintain in effect all provisions of the following Commission approved documents, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). These approved documents consist of information withheld from public disclosure pursuant to 10 CFR 2.790(d).

(1) "Security Plan for the Dresden Nuclear Power Station", dated November 19, 1977 as revised May 19, 1978, May 27, 1978, July 28, 1978 and February 19, 1979.

Am. 56
2/11/81

(2) "Dresden Nuclear Power Station Safeguards Contingency Plan", dated March 1980, as revised June 27, 1980, submitted pursuant to 10 CFR 73.40. The Contingency Plan shall be fully implemented, in accordance with 10 CFR 73.40(b), within 30 days of this approval by the Commission.

List of Figures

Insert A

Figure 3.5-1B Transient Linear Heat Generation Rate vs Nodal Exposure for all ANF Fuel

ID:8089N/1

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) - 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.~~

DELETE
INSERT AA

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.

Insert AA

K. <u>Steady State Linear Heat Generation Rate (SLHGR)</u> - The steady state linear heat generation rate limit protects against exceeding the fuel end-of-life steady state design criteria developed by Advanced Nuclear Fuels.

1.0 DEFINITIONS (Cont'd.)

AA. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alternations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

BB. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate the circuit in question.

CC. Surveillance Interval - Each surveillance requirement shall be performed within the specified surveillance interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval.

b. A total maximum combined interval time for any 3 consecutive intervals not to exceed 3.25 times the specified surveillance interval.

DD. Fraction of Rated Power (FRP) - The fraction of rated power is the ratio of core thermal power to rated thermal power of 2527 Mwth.

EE. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

~~FF. Maximum Fraction of Limiting Power Density (MFLPD) - The maximum fraction of limiting power density is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).~~

delete

Insert B-B

GG. Dose Equivalent I-131 - That concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

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Insert BB

FF. Fuel Design Limiting Ratio for Exxon Fuel (FDLRX) -
The fuel design limiting ratio for Exxon fuel is the limit used to assure that the fuel operates within the end-of-life steady state design criteria. FDRLX assures acceptable end-of-life conditions by, among other items, limiting the release of fission gas to the cladding plenum.

HH. Process Control Program (PCP) - Contains the sampling, analysis, and formulation determination by which solidification of radioactive wastes from liquid systems is assured.

II. Offsite Dose Calculation Manual (ODCM) - Contains the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitor alarm/trip setpoints.

JJ. Channel Functional Test (Radiation Monitor) - Shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

KK. Source Check - The qualitative assessment of instrument response when the sensor is exposed to a radioactive source.

LL. Member(s) of the Public - Shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MM. Rated Recirculation Pump Speed - is the recirculation pump speed that corresponds to rated core flow (98×10^6 lb/hr) when operating at rated thermal power (dual loop operation).

NN. Dual Loop Operation - reactor power operation with both recirculation pumps running.

OO. Single Loop Operation (SLO) - reactor power operation with one recirculation pump running.

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PP. <u>Transient Linear Heat Generation Rate (TLHGR)</u> - The transient linear heat generation rate limit protects against fuel centerline melting and 1% plastic cladding strain during transient conditions throughout the life of the fuel.

QQ. <u>Fuel Design Limiting Ratio for Centerline Melt (FDLRC)</u> - The fuel design limiting ratio for centerline melt is the limit used to assure that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of rated thermal power.

RR. <u>Linear Heat Generation Rate (LHGR)</u> - The linear heat generation rate is the operating fuel pin power level.

1.1 SAFETY LIMIT

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 greater than 800 psig and Core Flow greater than 10% of Rated.

The existence of a minimum critical power ratio (MCPR) less than 1.06 for GE ~~8x8R~~ fuel, or less than 1.05 ANF for ~~ENC or GE 8x8~~ fuel, shall constitute violation of the MCPR fuel cladding integrity safety limit.

When in Single Loop Operation, the MCPR safety limit shall be increased by ~~0.03.~~ 0.01

2.1 LIMITING SAFETY SYSTEM SETTING

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 less than or equal to [$.58W_D$ + 62] during Dual Loop Operation or S less than or equal to [$.58W_D$ + 58.5] during Single Loop Operation

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

S - setting in percent of rated thermal power.

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1.1 SAFETY LIMIT (Cont'd.)

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

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

In the event of operation of any fuel assembly with a ~~maximum fraction of limiting power density (MFLPD)~~ fuel design limiting ratio for centerline melt (FDLRC) greater than ~~the~~ 1.0 ~~fraction of rated power (FRP)~~, the setting shall be modified as follows:

Where: S is less than or equal to ~~FRP/MFLPD~~ 1/FDLRC (.58W_D + 62) ~~[FRP/MFLPD]~~ during Dual Loop Operation or (.58 W_D + 58.5) ~~[FRP/MFLPD]~~ 1/FDLRC during Single Loop Operation

~~FRP = fraction of rated thermal power~~

~~MFLPD = Maximum Fraction of the Limiting Power Density for all fuel types~~

~~The value of FDLRC shall be set equal to 1.0 unless the actual operating value is greater than 1.0, in which case the actual operating value will be used~~

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

This adjustment may also be performed by increasing the APRM gain by ~~the inverse ratio, MFLPD/FRP~~ FDLRC, which accomplishes the same degree of protection as reducing the trip setting by ~~FRP/MFLPD.~~ 1/FDLRC

2. **APRM Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)**

1.1 **SAFETY LIMIT** (Cont'd.)

B. Core Thermal Power Limit (Reactor Pressure is less than or equal to 800 psig)

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

2.1 **LIMITING SAFETY SYSTEM SETTING** (Cont'd.)

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

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 is less than or equal to [$.58W_D$ + 50] during Dual Loop Operation or S is less than or equal to [$.58\ W_D$ + 46.5] during Single Loop Operation.

The definitions used above for the APRM scram trip apply.

In the event of operation of any fuel assembly with a ~~maximum fraction limiting power density (MFLPD)~~ fuel design limiting ratio for centerline melt (FDLRC) greater than ~~the fraction of rated power (FRP)~~ 1.0, the setting shall be modified as follows:

S is less than or equal to ($.58W_D$ + 50)/~~(FRP/MFLPD)~~ FDLRC during Dual Loop Operation or S is less than or equal to (.58 W_D + 46.5)/~~(FRP/ MFLPD)~~ FDLRC during Single Loop Operation

The definitions used above for the APRM scram trip apply.

The ~~ratio~~ value of ~~FRP to MFLPD~~ FDLRC shall be set equal to 1.0 unless the actual operating value is ~~less~~ greater than 1.0. In which case the actual operating value will be used.

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1.1 SAFETY LIMIT (Cont'd.)

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.

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

2.1 LIMITING SAFETY SYSTEM SETTING (Cont'd.)

The adjustment may also be performed by increasing the APRM gain by ~~the inverse ratio, MFLPD/FRP~~ FDLRC, which accomplishes the same degree of protection as reducing the trip setting by ~~FRP/MFLPD.~~ 1/FDLRC

C. Reactor low water level scram setting shall be greater than or equal to 144" above the top of the active fuel at normal operating conditions.

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

D. Reactor low water level ECCS initiation shall be 84"(+4", minus 0") above the top of the active fuel at normal operating conditions.

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

1.1 SAFETY LIMIT BASES (Cont'd.)

power ratio (CPR) which is the ratio of the bundle power which would produce the 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 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.

During Single Loop Operation, the MCPR safety limit is increased by ~~0.03~~ to conservatively account for increased uncertainties in the core flow and TIP measurements.

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1.1 SAFETY LIMIT BASES (Cont'd.)

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 1/FDLRC ~~the ratio of FRP/MFLPD~~. Specification 3.5.J establishes the maximum value of LHGR which cannot be exceeded during steady power operation for any fuel types.

For fuel fabricated by Advanced Nuclear Fuels Corporation (ANF), ANF ~~Exxon Nuclear Company (ENC), ENC~~ has performed fuel design analysis which demonstrate that fuel centerline melting point will not be reached during transient overpower condition throughout the design life of the fuel provided FDLRC is monitored. The analysis has also shown that the design criteria of 1.0% uniform cladding strain will not be exceeded during both steady state and transient operation throughout the fuel design life provided FDLRC is monitored.

B. Core Thermal Power Limit (Reactor Pressure less than 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of $28x10^3$ lbs/hr. bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow

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2.1 LIMITING SAFETY SYSTEM SETTING BASES (Cont'd.)

At times it may be necessary to operate with one reactor coolant recirculation pump out of service. During Single Loop Operation, the normal drive flow relationship during Dual Loop Operation is altered. This is the result of reverse flow through the idle loop jet pumps when the active loop recirculation pump speed is above 40% of rated. Some of the active loop flow is then diverted from the core and backflows through the idle loop jet pumps; hence, the core receives less flow than would be predicted based upon the Dual Loop drive flow-to-core flow relationship. If the APRM flow biased trip settings were not altered for Single Loop Operation, the new drive flow to core flow relationship would nonconservatively result in flow biased trips occurring at neutron fluxes higher than normal for a given core flow.

limit violated

The scram trip setting must be adjusted to ensure that the LHGR transient ~~peak~~ is not ~~increased~~ for any ~~combination of Maximum Fraction of Limiting Power Density (MFLPD) and reactor core thermal power.~~ The scram setting is ~~adjusted~~ decreased in accordance with the formula in specification 2.1.A.1 when the ~~MFLPD~~ FDLRC is greater than ~~the fraction of rated power (FRP).~~ 1.0.

power distribution. This is expressed as the Fuel Design Limiting Ratio for Centerline melt (FDLRC)

The adjustment may also be accomplished by increasing the APRM gain by ~~the reciprocal of FRP/MFLPD.~~ 1/FDLRC This provides the same degree of protection as reducing the trip setting by ~~FRP/MFLPD~~ FDLRC 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 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

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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 worse case MCPR which could occur during steady-state operation is at 108% of rated thermal power during dual loop operation or 104.5 percent during single loop operation because of the APRM rod block trip setting. As with the APRM flow biased scram, the reduced setpoint during single loop operation accounts for possible reverse flow in the idle loop jet pumps. 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 any fuel type exceeds the fraction of rated power~~, thus preserving the APRM rod block safety margin.

fuel design limiting ratio for centerline melt (FDLRC) for any fuel assembly exceeds 1.0,

C. Reactor Low Water Level Scram - The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained. The scram setpoint is based on normal operating temperature and pressure conditions because the level instrumentation is density compensated.

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

3.1 LIMITING CONDITIONS FOR OPERATION

REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

A. Reactor Protection System

1. 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.

~~2. If during operation, the maximum fraction of limiting power density exceeds the fraction of rated power when operating above 25% rated thermal power, either:~~

~~delete~~

2. If during operation, the fuel design limiting ratio for centerline melt (FDLRC) for any fuel assembly exceeds 1.0 when operating above 25% rated thermal power, either:

4.1 SURVEILLANCE REQUIREMENTS

REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

A. Reactor Protection System

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

~~2. Daily during reactor power operation above 25% rated thermal power, the core power distribution shall be checked for:~~

~~delete~~

2. Daily during reactor power operation above 25% rated thermal power, the core power distribution shall be checked for:

3.1 LIMITING CONDITIONS FOR OPERATION (Cont'd.)

a. The APRM scram and rod block 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.

b. The power distribution shall be changed such that the ~~maximum fraction of limiting power density no longer exceeds the fraction of rated power.~~ fuel design limiting ratio for centerline melt (FDLRC) for any fuel assembly no longer exceeds 1.0.

3. Two RPS electric power monitoring channels for each inservice RPS MG set or alternate source shall be OPERABLE at all times.

4.1 SURVEILLANCE REQUIREMENTS (Cont'd.)

a. ~~Maximum fraction of limiting power density (MFLPD) and compared with the fraction of rated power (FRP)~~. Maximum fuel design limiting ratio for centerline melt (FDLRC)

b. Deleted.

3. The RPS power monitoring system instrumentation shall be determined OPERABLE:

a. At least once per 6 months by performing a CHANNEL FUNCTIONAL TEST, and

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4.1 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

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 Volume dP and Thermal Switches, 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~~ FDLRC 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~~ FDLRC is adequate.

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TABLE 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum No. of Operable Inst. Channels Per Trip System (1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias) (7)	
	Dual Loop Operation	Less than or equal to (.58 W_D plus 50) ~~(FRP/MFLPD)~~ / FDLRC (See Note 2)
	Single Loop Operation	Less than or equal to (.58 W_D plus 46.5) ~~(FRP/MFLPD)~~ / FDLRC (See Note 2)
1	APRM upscale (refuel and Startup/Hot Standby mode)	Less than or equal to 12/125 full scale
2	APRM downscale (7)	Greater than or equal to 3/125 full scale
1	Rod block monitor upscale (flow bias) (7) ~~(8)~~	
	Dual Loop Operation	Less than or equal to (.65 W_D plus 45) (See Note 2)
	Single Loop Operation	Less than or equal to (.65 W_D plus 41) (See Note 2)
1	Rod block monitor downscale (7)	Greater than or equal to 5/125 full scale
3	IRM downscale (3)	Greater than or equal to 5/125 full scale
3	IRM upscale	Less than or equal to 108/125 full scale
3	IRM detector not fully inserted in the core	N/A
2 (5)	SRM detector not in startup position	(4)
2 (5) (6)	SRM upscale	Less than or equal to 10^5 counts/sec.
1 (per bank)	Scram discharge volume water level - high	(LT/E) 26 inches above the bottom of the instrument volume

Notes: (See Next Page)

TABLE 3.2.3 (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. A 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 per bank.

2. W_D percent of drive flow required to produce a rated core flow of 98 Mlb/hr. ~~MFLPD = highest value of FLPD.~~
   FDLRC = fuel design limiting ratio for centerline melt.

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

4. This function may be bypassed when the count rate is greater than or equal to 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 test at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

~~8. Instrument trip level setting shall not exceed its value at rated flow.~~

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3.3 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

D. Control Rod Accumulators

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

1. Inoperable accumulator,
2. Directional control valve electrically disarmed while in a non-fully inserted position.

4.3 SURVEILLANCE REQUIREMENT (Cont'd.)

2. At 16 week intervals, at least 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% or more of the control rod drives have been tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

delete

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.

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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 pattern, 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, -3.5.L the negative reactivity insertion rates associated with the ~~observed~~ scram performance ~~(as adjusted for statistical variation in the observed data)~~ as specified in Specification 3.3.C. 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 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve 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

3.3 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The bounding value described above was used in the transient analysis

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 analyses, 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 re-determination of thermal margin requirements is undertaken (as required by Specification 3.5.K) unless it can be shown that the number of individual drives falling 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.~~

delete

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.

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3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

D. Automatic Pressure Relief Subsystems

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

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~~

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

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

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. 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.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

~~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.~~

~~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.~~

see insert 3A

4. If the requirements of 3.5.D. cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to below 90 psig within 24 hours.

E. Isolation Condenser System

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

~~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.~~

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

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reactor operation is permissible only during the succeeding seven (7) days provided that during such time the HPCI subsystem is operable. If the following MAPLHGR reduction factors (multipliers) are applied to Figure 3.5-1, the Automatic Pressure Relief Subsystem of ECCS shall be considered operable: (1) 0.89 for 8x8 fuel, or (2) 0.76 for 9x9 fuel.

3. From and after the date that two relief valves are found or made to be inoperable, reactor operation is permissible only during the succeeding seven days provided that during such time the HPCI subsystem is operable and the multipliers specified in 3.5.D.2 are applied.

3. Whenever HPCI is required to be operable in accordance with 3.5.D.3, HPCI shall be tested to demonstrate operaability immediately.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

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~~ ANF fuel at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5-1. For operation during Single Loop Operation, the values of Figure 3.5-1 shall be decreased ~~to 90% of the original value~~ by a multiplicative factor of 0.91. <Insert C> 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.

J. LOCAL STEADY STATE LHGR

During steady state power operation above 25% rated thermal power, the linear heat generation rate (LHGR) of any rod in any fuel assembly

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

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~~ ANF fuel shall be determined daily during reactor operation at greater than or equal to 25% rated thermal power.

J. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

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If, concurrently, one Automatic Pressure Relief Subsystem relief valve is out-of-service, the values of Figure 3.5-1 shall be decreased by a multiplicative factor of 0.89 for 8x8 fuel and 0.76 for 9x9 fuel.

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3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

at any axial location shall not exceed its maximum steady state LHGR (SLHGR) value shown in Figure 3.5-1A (consists of three curves). That is, the Fuel Design Limiting Ratio for Exxon Fuel (FDLRX) shall not be greater than 1.0 where FDLRX = $\frac{LHGR}{SLHGR}$

Figure 3.5-1A depicts the Steady State LHGR values for ~~Exxon~~ ANF 8x8 and 9x9 fuel as a function of nodal exposure and for GE 8x8 fuel as a constant design value of 13.4 Kw/ft.

If at any time during operation above 25% rated thermal power it is determined by normal surveillance that ~~the limiting value for LHGR for any fuel assembly is being exceeded~~ FDLRX for any fuel assembly exceeds 1.0, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the ~~LHGR~~ FDLRX 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.

< Insert D >

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K. Local Transient LHGR

At any time during power operation, above 25% rated thermal power the Fuel Design Limiting Ratio for Centerline melt (FDLRC) shall not be greater than 1.0, where

For ANF fuel:

$$FDLRC = \frac{LHGR * 1.2}{TLHGR * FRP}$$

Figure 3.5-1B depicts the TLHGR values for ANF 8x8 and 9x9 fuel as a function of nodal exposure.

If during operation, the FDLRC exceeds 1.0 when operating above 25% rated thermal power, either:

a. The APRM scram and rod block 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.

b. The power distribution shall be ~~changed~~ changed such that the FDLRC no longer exceeds 1.0.

K. Transient Linear Heat Generation Rate (LHGR)

The fuel design limiting ratio for centerline melt (FDLRC) shall be checked daily during reactor operation at greater than or equal to 25% rated thermal power.

For GE fuel:

$$FDLRC = \frac{LHGR}{FRP * SLHGR}$$

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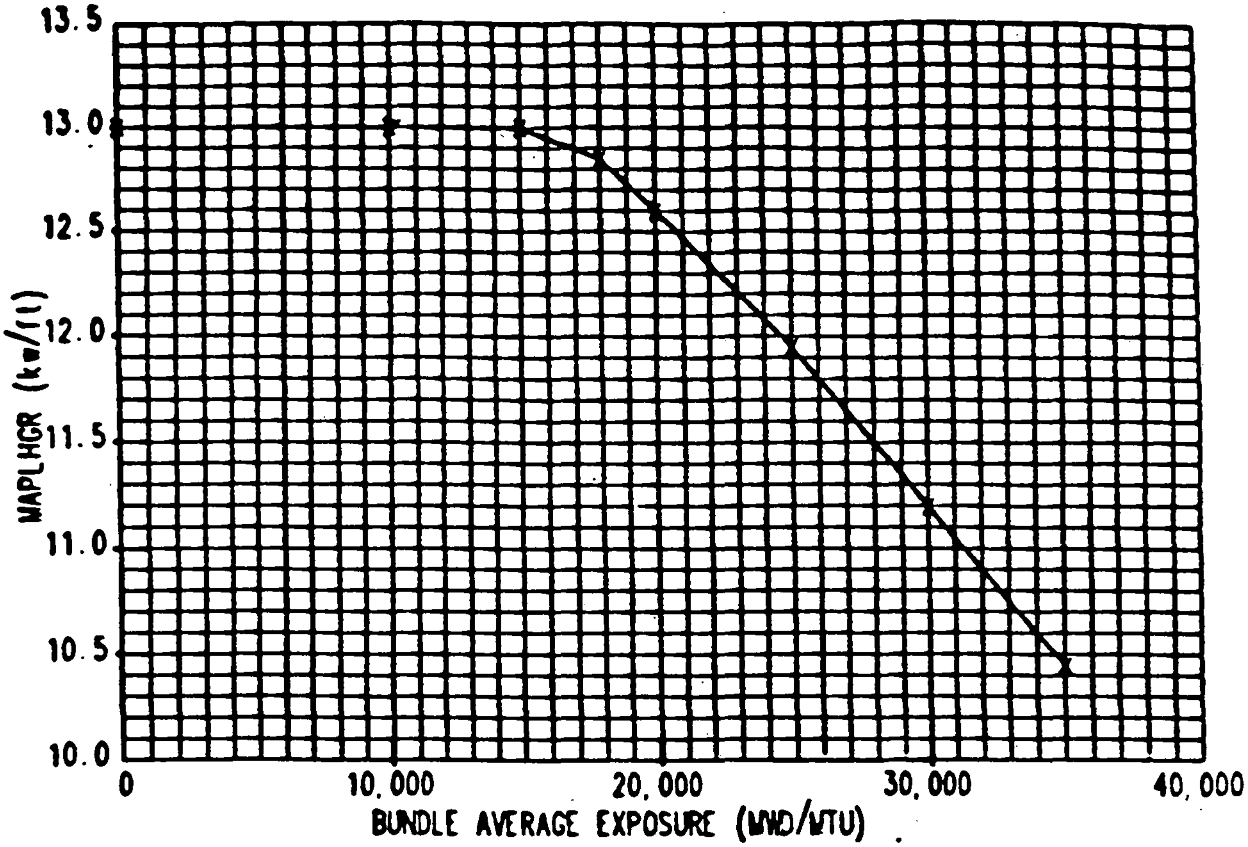
MAPLHGR LIMIT VS. BUNDLE AVERAGE EXPOSURE
ANF ~~ENC~~ 8x8 FUEL



The above graph is based on the following MAPLHGR summary for ANF ~~ENC~~ 8x8 fuel design:

Bundle Average Exposure (MWD/MTU)	MAPLHGR Limit (kw/ft)
0	13.0
10,000	13.0
15,000	13.0
18,000	12.85
20,000	12.60
25,000	11.95
30,000	11.20
35,000	10.45

Figure 3.5-1
(Sheet 1 of ~~6~~ 3)

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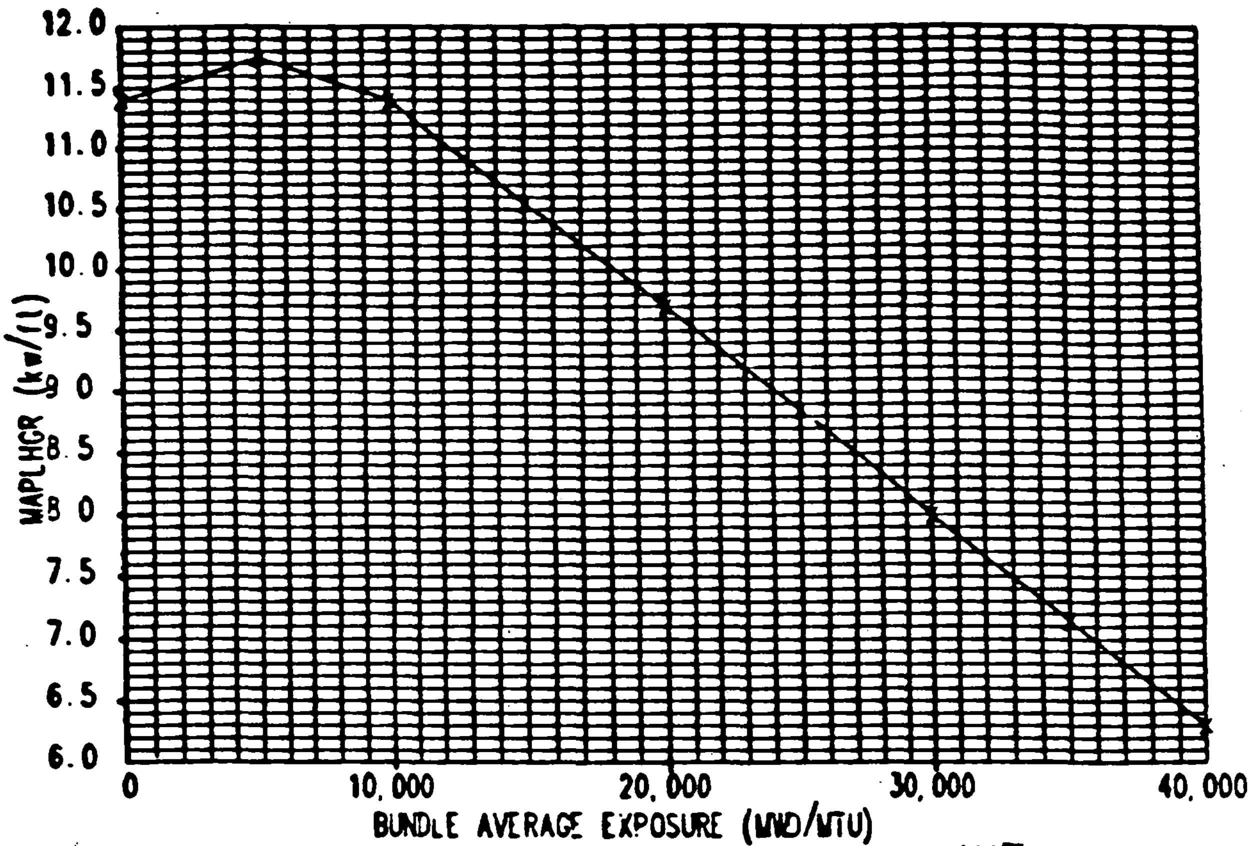
MAPLHGR LIMIT VS. BUNDLE AVERAGE EXPOSURE

ANF ~~ENC~~ 9x9 FUEL



The above graph is based on the following MAPLHGR summary for ANF ~~ENC~~ 9x9 fuel design:

Bundle Average Exposure (MWD/MTU)	MAPLHGR Limit (kw/ft)
0	11.40
5,000	11.75
10,000	11.40
15,000	10.55
20,000	9.70
25,000	8.85
30,000	8.00
35,000	7.15
40,000	6.30

Figure 3.5-1
(Sheet 2 of ~~6~~ 3)

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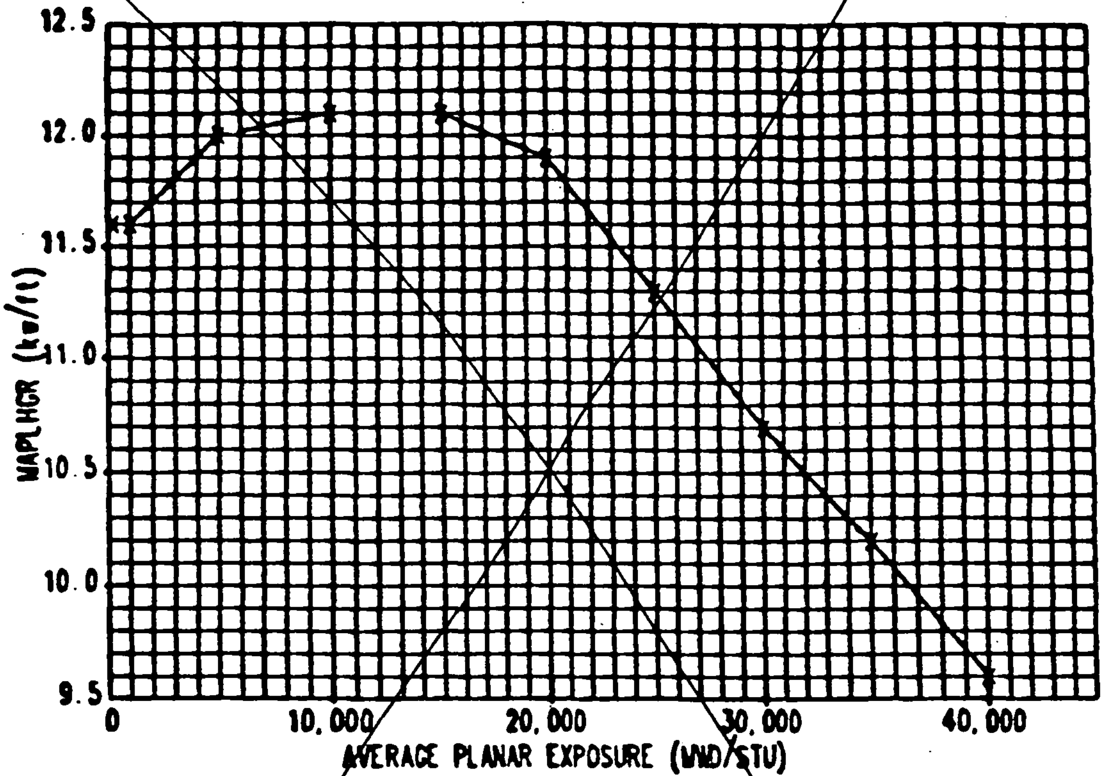
MAPLHGR LIMIT VS. AVERAGE PLANAR EXPOSURE
GE FUEL TYPE P8DRB265L



The above graph is based on the following MAPLHGR summary for GE fuel type P8DRB265L.

Average Planar Exposure (MWD/STU)	MAPLHGR Limit (kw/ft)
200	11.6
1,000	11.6
5,000	12.0
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

Figure 3.5-1
(Sheet 3 of 6)

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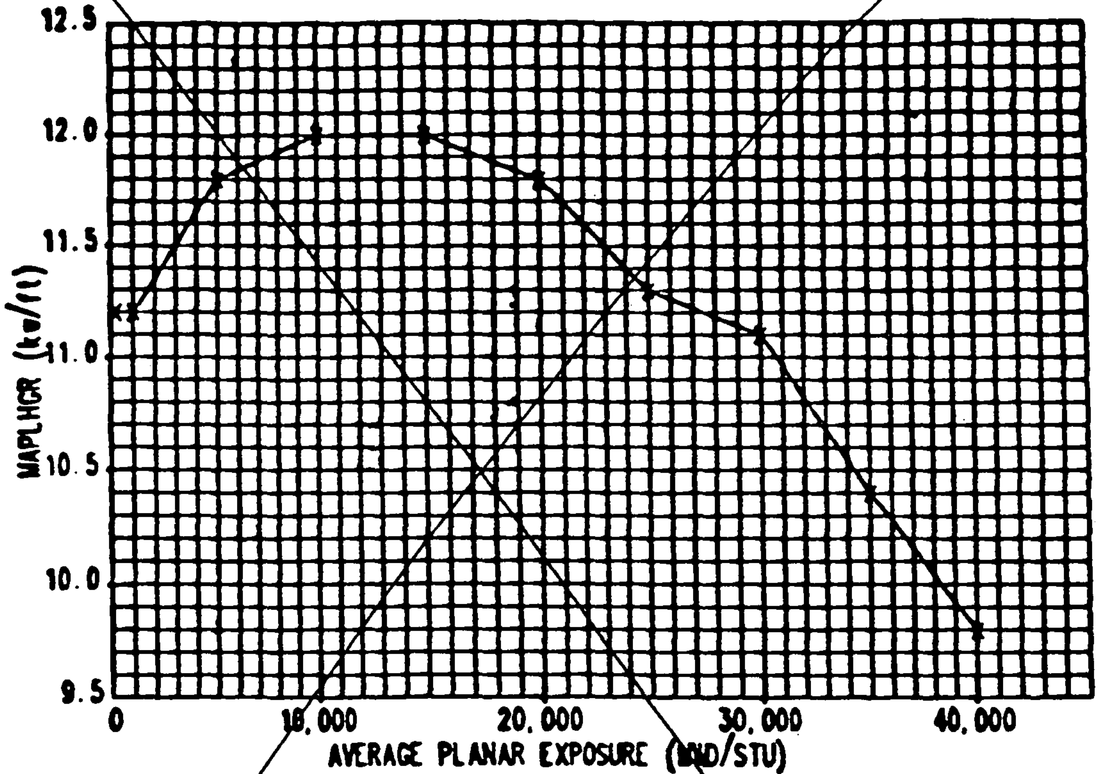
MAPLHGR LIMIT VS. AVERAGE PLANAR EXPOSURE
GE FUEL TYPE P8DRB282



The above graph is based on the following MAPLHGR summary for GE fuel type P8DRB282.

Average Planar Exposure (MWD/STU)	MAPLHGR Limit (kw/ft)
200	11.2
1,000	11.2
5,000	11.8
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.3
30,000	11.1
35,000	10.4
40,000	9.8

Figure 3.5-1
(Sheet 4 of 6)

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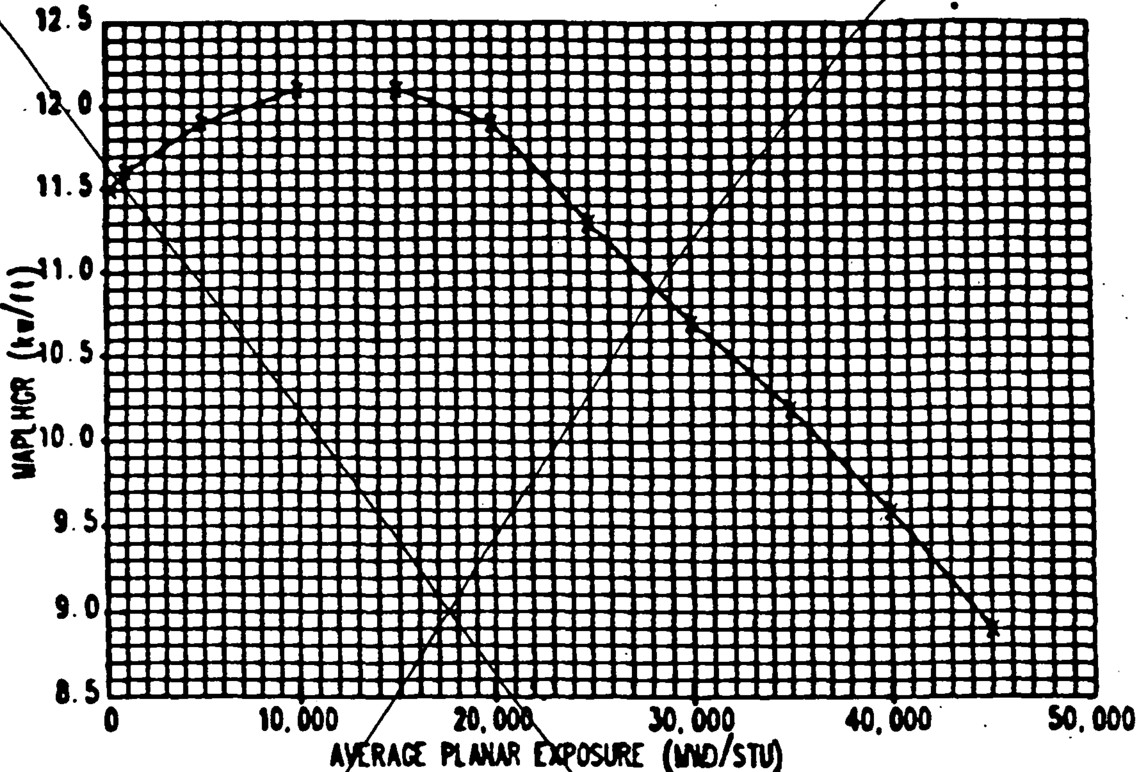
MAPLHGR LIMIT VS. AVERAGE PLANAR EXPOSURE
GE FUEL TYPE P8DRB265H



The above graph is based on the following MAPLHGR summary for GE fuel type P8DRB265H

Average Planar Exposure (MWD/STU)	MAPLHGR Limit (kw/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6
45,000	8.9

Figure 3.5-1
(Sheet 5 of 6)

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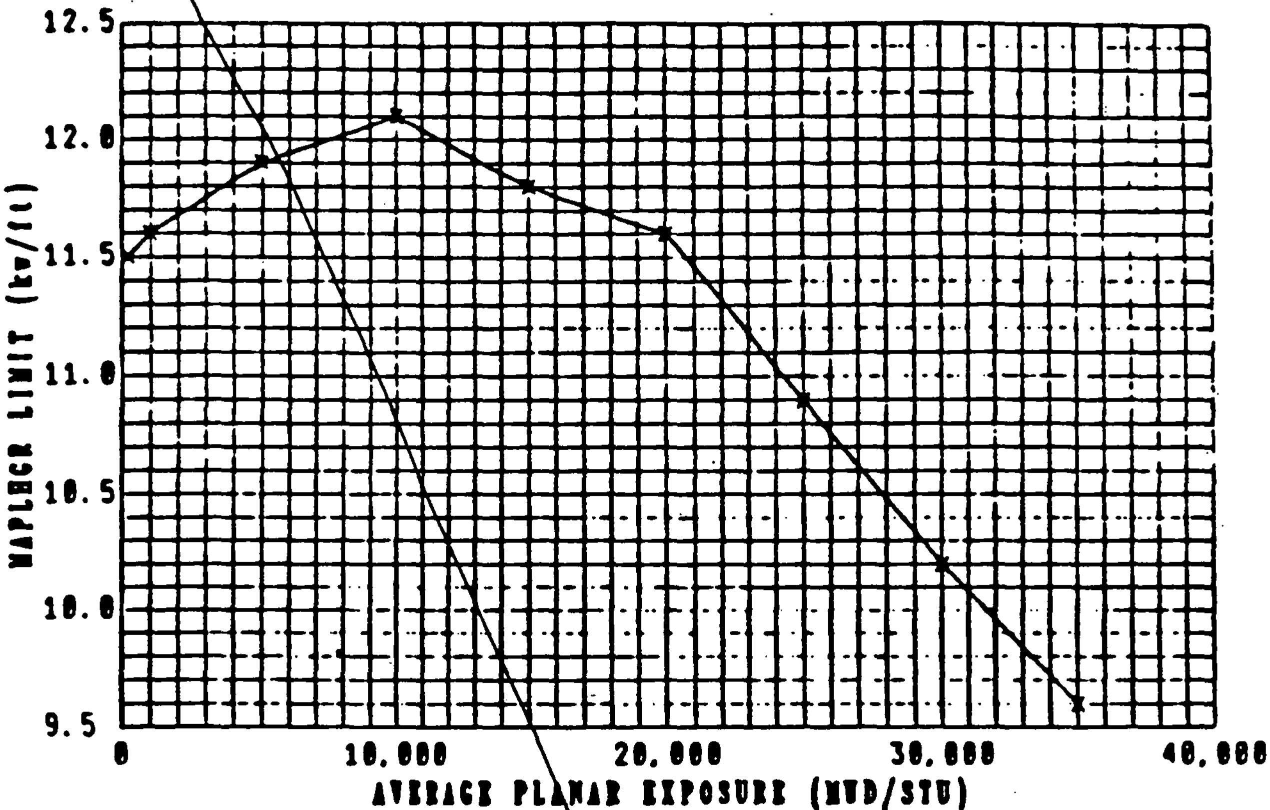
MAPLHGR VS. Average Planar Exposure
GE 8X8 LTA - Bundle LY5458



The above graph is based on the following MAPLHGR summary for the GE LTA, bundle LY5458:

Average Planar Exposure (MWD/STU)	MAPLHGR Limit (kw/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	11.8
20,000	11.6
25,000	10.9
30,000	10.2
35,000	9.6

replace with Insert JJ

Figure 3.5-1
(Sheet ~~6~~ 3 of ~~6~~ 3)

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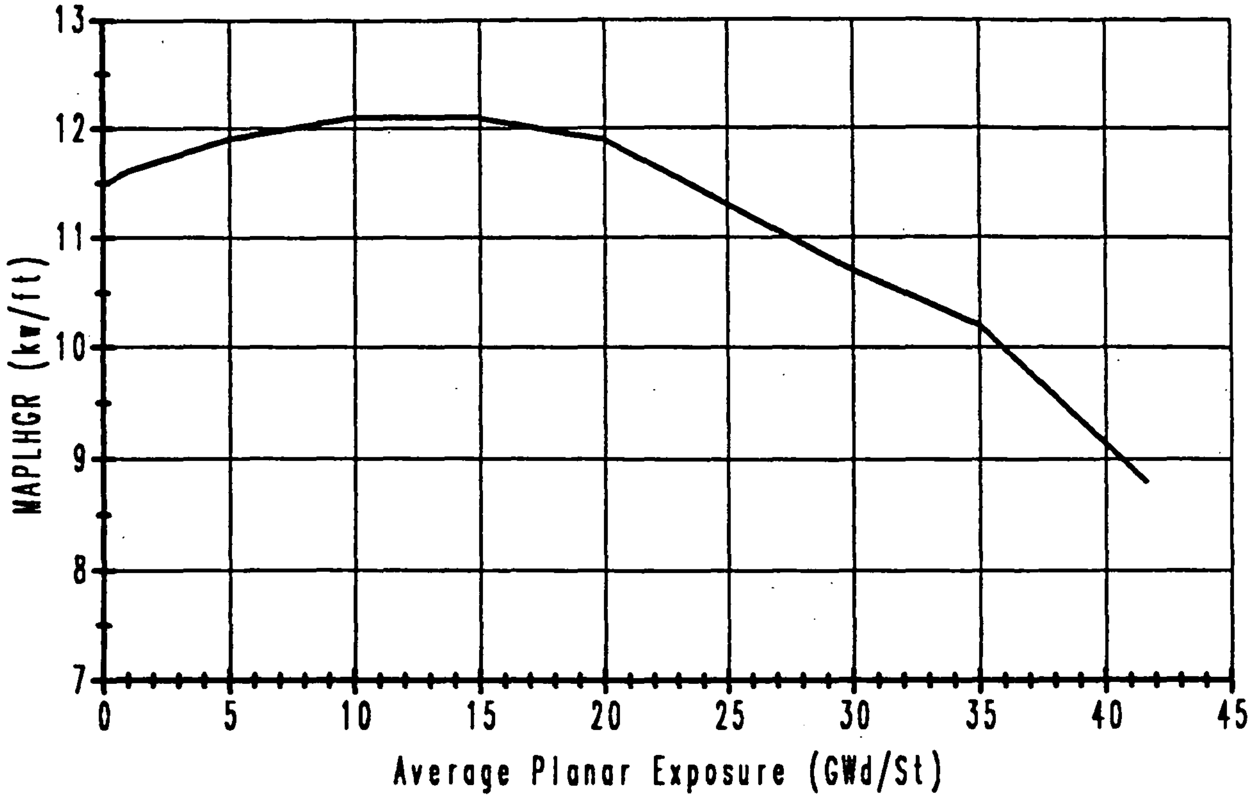
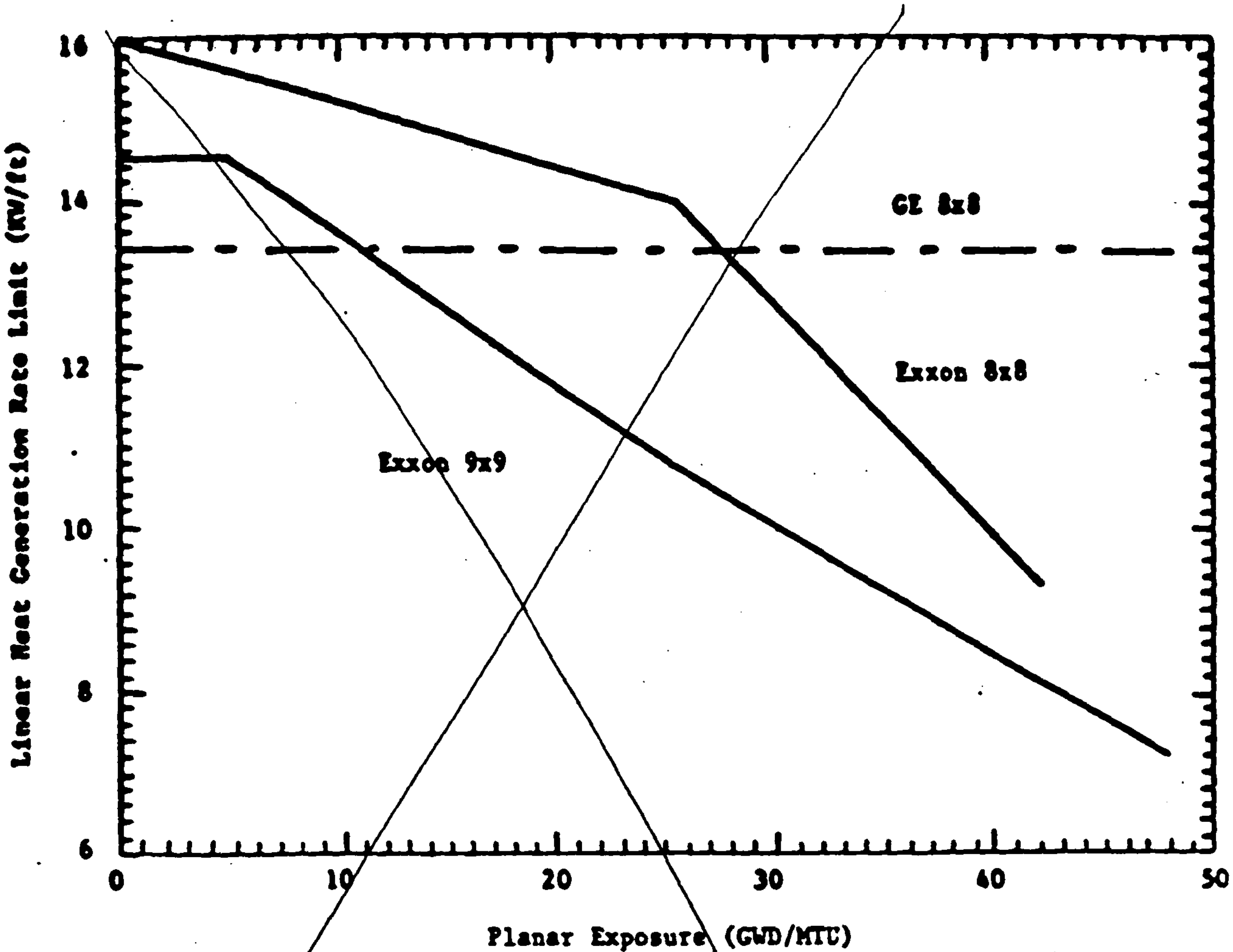
MAPLHGR Vs. Average Planar Exposure
GE 8x8 LTAs



The above graph is based on the following bounding MAPLHGR summary for the GE LTAs.

Average Planar Exposure (GWd/St)	MAPLHGR (KW/FT)
0.2	11.5
1.0	11.6
5.0	11.9
10.0	12.1
15.0	12.1
20.0	11.9
25.0	11.3
30.0	10.7
35.0	10.2
41.6	8.8

Figure 3.5-1 (Sheet 3 of 3)



Exxon 8x8 Fuel	
Exposure	LHGR
0.00	16.00
25.40	14.10
42.00	9.30

Exxon 9x9 Fuel	
Exposure	LHGR
0.00	14.50
5.00	14.50
25.20	10.80
48.00	7.20

Figure 3.5-1A

LINEAR HEAT GENERATION RATE VS.
NODAL EXPOSURE

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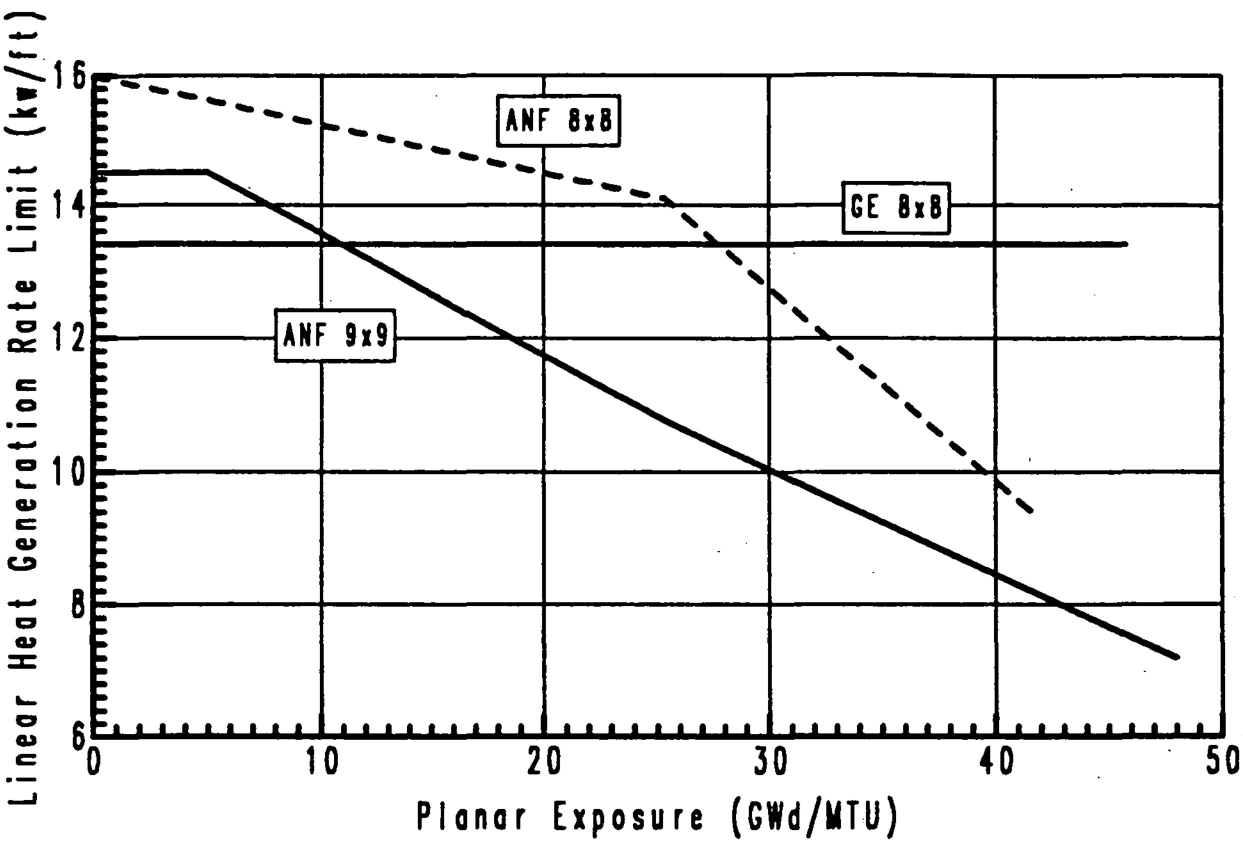
STEADY STATE LINEAR HEAT GENERATION RATE LIMIT (SLHGR) VS. NODAL EXPOSURE



GE 8x8 Fuel Exposure	GE 8x8 Fuel LHGR	ANF 8x8 Fuel Exposure	ANF 8x8 Fuel LHGR	ANF 9x9 Fuel Exposure	ANF 9x9 Fuel LHGR
0.0	13.4	0.0	16.0	0.0	14.5
45.8	13.4	25.4	14.1	5.0	14.5
		42.0	9.3	25.2	10.8
				48.0	7.2

Figure 3.5-1A

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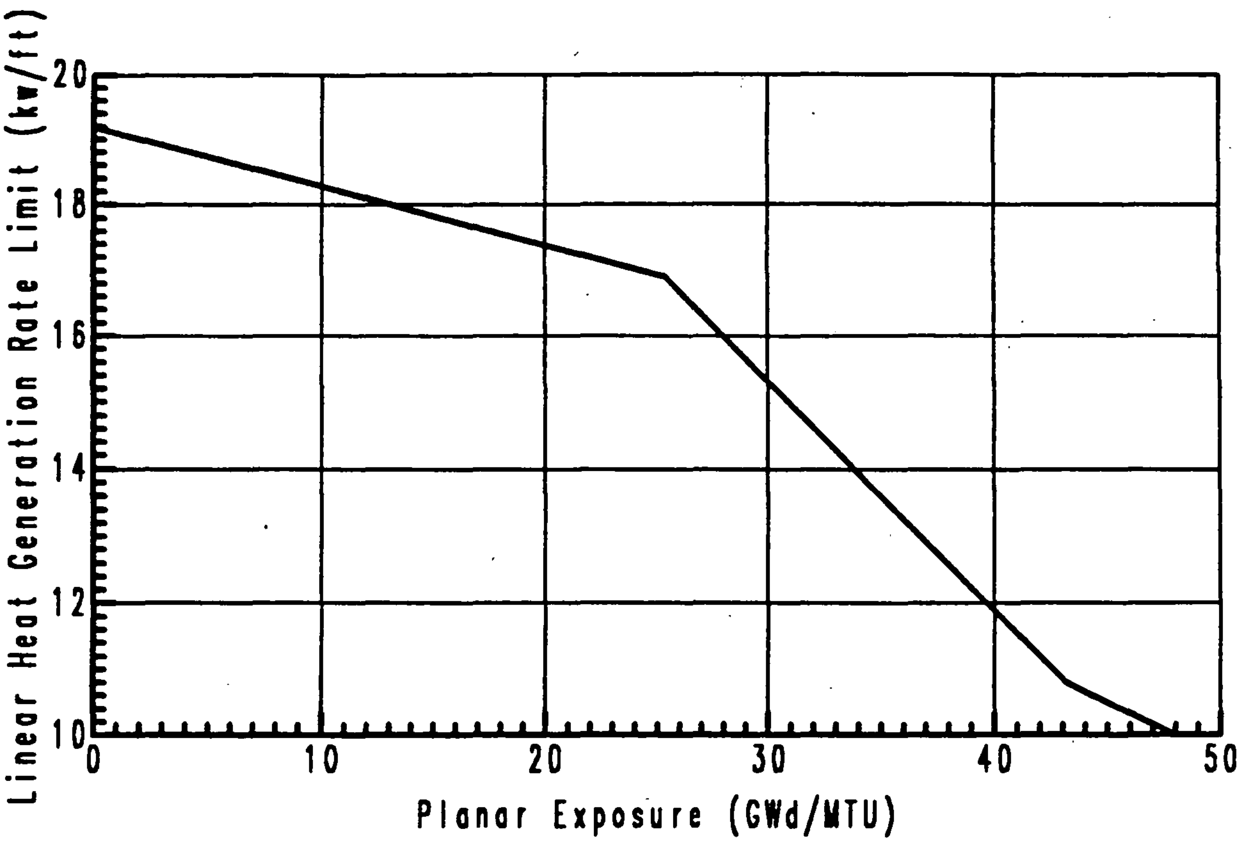
TRANSIENT LINEAR HEAT GENERATION RATE LIMIT (TLHGR) VS. NODAL EXPOSURE FOR ALL ANF FUEL



Exposure	LHGR
0.0	19.2
25.4	16.9
43.2	10.8
48.0	10.0

Figure 3.5-1B

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

~~K~~. L Minimum Critical Power Ratio (MCPR)

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

1.39. ~~1.32 for Exxon 9X9 fuel~~

~~1.31 for GE and Exxon 8x8 fuel~~

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 is the greatest of the following:
   a. The above rated core flow value;
   b. The value from Figure 3.5-2 sheet 1; or
   c. The ~~interpolated~~ value from Figure 3.5-2 sheet~~s~~ 2 ~~and 3.~~

3. During Single Loop Operation, all the rated flow MCPR operating limits shall be increased by an additive factor of ~~0.03~~. 0.01

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

~~K~~. L Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at greater than or equal to 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.

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3.5 <u>LIMITING CONDITION FOR OPERATION</u>
(Cont'd.)

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 Specification 3.3.C for all operable control rods exceeds 2.77 seconds, the MCPR operating limit shall be increased by adding the amount equal to [0.238T - 0.66] where T equals the average 90% scram insertion time for the most recent half-core or full core surveillance data from Specification 4.3.C. Consequentially, the Automatic Flow Control MCPR Operating Limit must also be evaluated in accordance with Specification 3.5.K.2.~~

4.5 <u>SURVEILLANCE REQUIREMENT</u>
(Cont'd.)

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3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

M ~~L~~. Condensate Pump Room Flood Protection

1. The system is installed to prevent or mitigate the consequences of flooding of the condensate pump room shall be operable prior to startup of the reactor.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

M ~~L~~. Condensate Pump Room Flood Protection

1. The following surveillance requirements shall be observed to assure that the condensate pump room flood protection is operable.

   a. The testable penetrations through the walls of CCSW pump vaults shall be checked during each operating cycle by pressurizing to 15 plus or minus 2 psig and checking for leaks using a soap bubble solution. The criteria for acceptance should be no visible leakage through the soap bubble solution. The bulkhead door shall be checked during each operating cycle by hydrostatically testing the door at 15 plus or minus 2 psig and checking to verify that leakage around the door is less than one gallon per hour.

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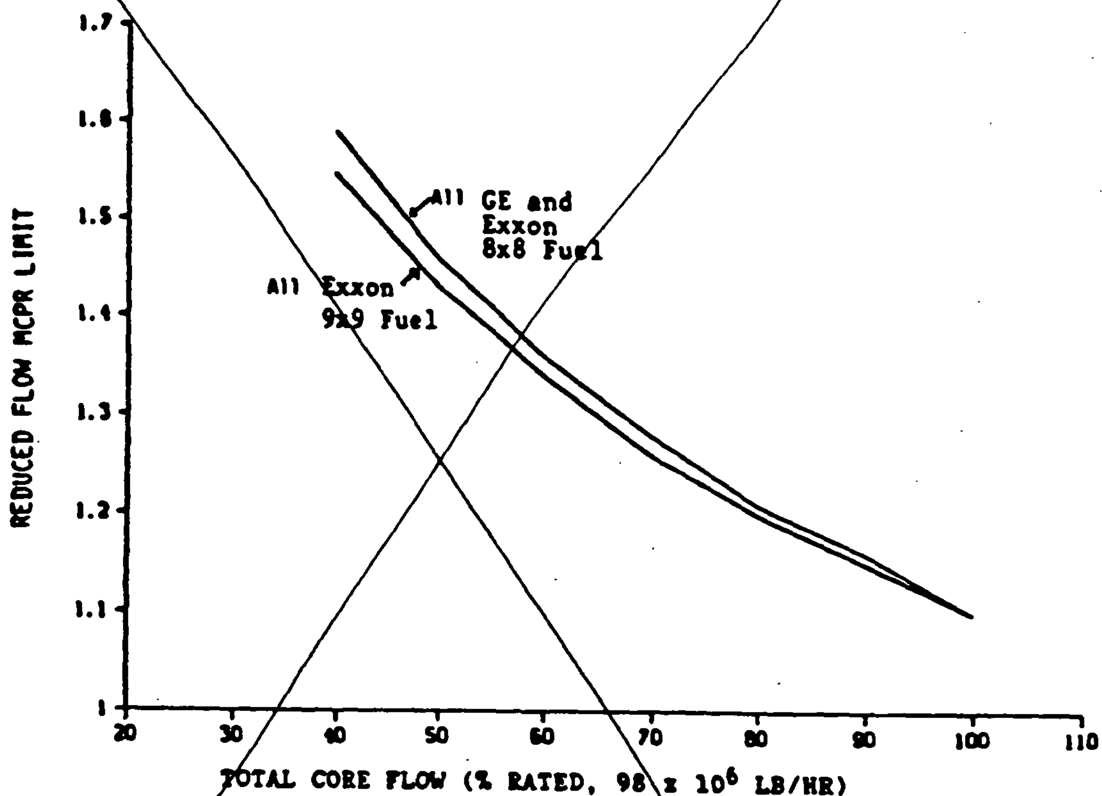

The above curves are based on the following MCPR Limit summary for reduced Total Core Flow:

Total Core Flow (% Rated)	MCPR Limit GE and Exxon 8x8	MCPR Limit Exxon 9x9
100	1.10	1.10
90	1.16	1.15
80	1.21	1.20
70	1.28	1.26
60	1.36	1.34
50	1.46	1.43
40	1.59	1.55

Figure 3.5-2 (Sheet 1 of 3)
MCPR Limit for reduced Total Core Flow

replace with
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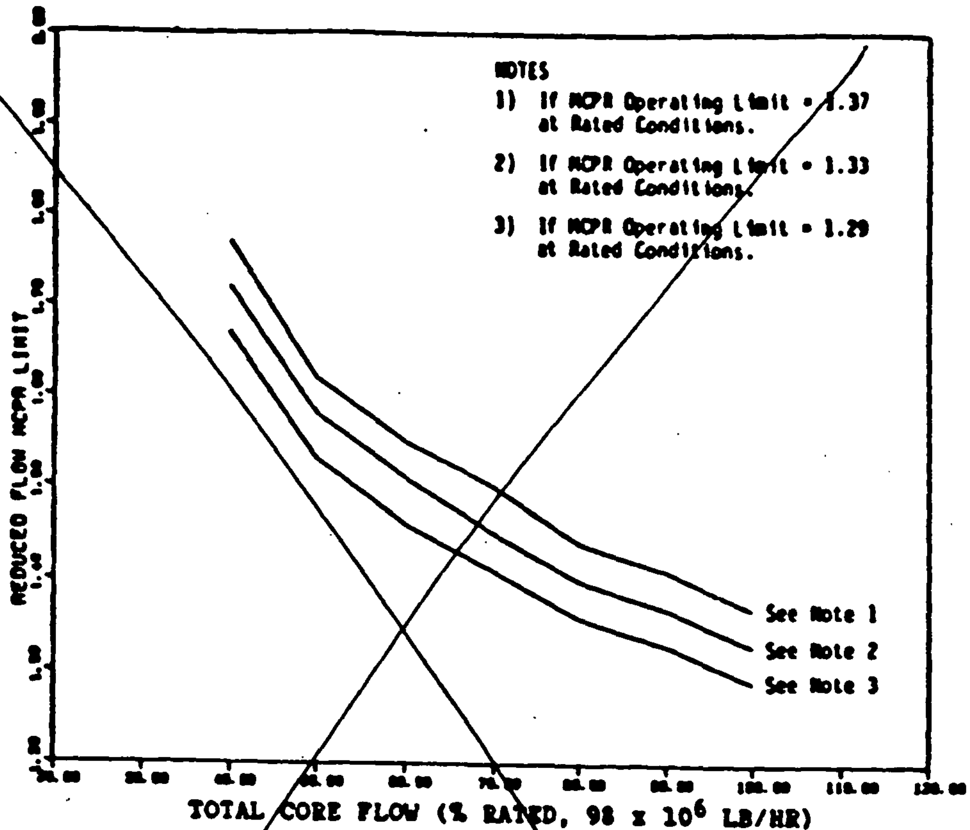

The above GE and Exxon 8x8 curves are based on the following MCPR operating limit summary for Automatic Flow Control:

Total Core Flow (% Rated)	MCPR Operating Limit for GE and Exxon 8x8 fuel* 1.29	1.33	1.37
100	1.29	1.33	1.37
90	1.33	1.37	1.41
80	1.36	1.40	1.44
70	1.41	1.45	1.50
60	1.46	1.51	1.55
50	1.53	1.58	1.62
40	1.67	1.72	1.77

* Column headers are MCPR Operating Limits at rated flow.

Figure 3.5-2 (Sheet 2 of 3)
GE and Exxon 8x8 MCPR Operating Limit For Automatic Flow Control

(replace with insert N)

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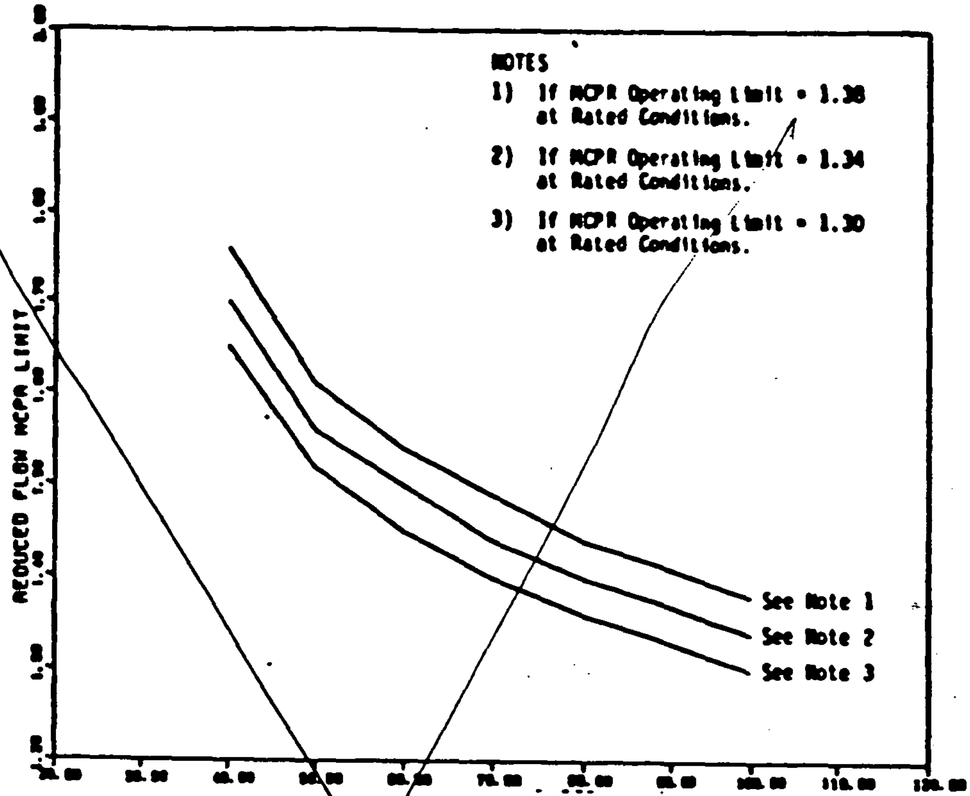

TOTAL CORE FLOW (% RATED, 98 x 10^6 LB/HR)

The above Exxon 9x9 curves are based on the following MCPR operating limit summary for Automatic Flow Control:

Total Core Flow (% Rated)	MCPR Operating Limit for Exxon 9x9 fuel* 1.30	1.34	1.38
100	1.30	1.34	1.38
90	1.33	1.37	1.41
80	1.36	1.40	1.44
70	1.40	1.44	1.49
60	1.45	1.50	1.54
50	1.52	1.56	1.61
40	1.65	1.70	1.76

* Column headers are MCPR Operating Limits at rated flow.

Figure 3.5-2 (Sheet 3 of 3)
Exxon 9x9 MCPR Operating Limit For Automatic Flow Control

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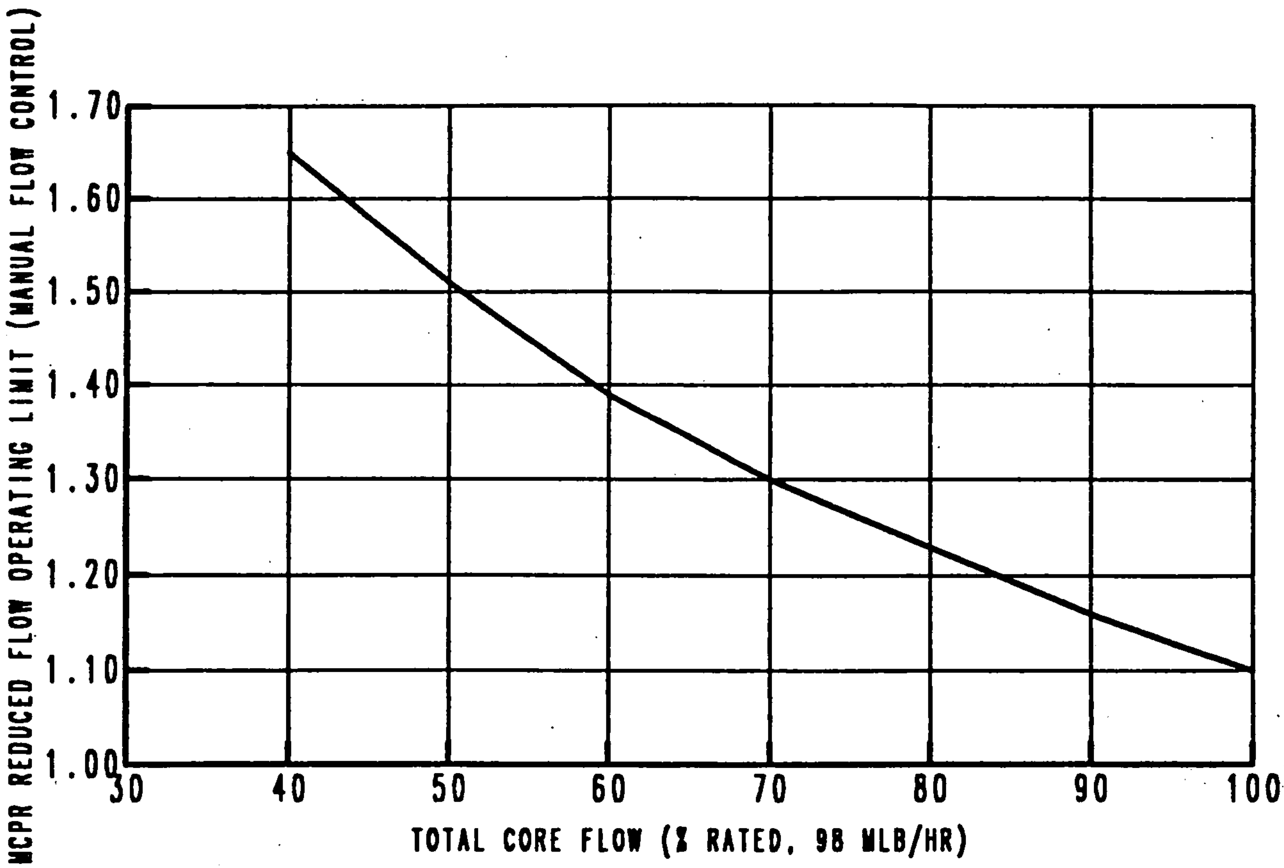
MCPR Limit for Reduced Total Core Flow



The above curve is based on the following MCPR operating limit summary for reduced core flow and all fuel types:

Total Core Flow (% Rated)	MCPR Operating Limit
100	1.10
90	1.16
80	1.23
70	1.30
60	1.39
50	1.51
40	1.65

Figure 3.5-2 (Sheet 1 of 2)

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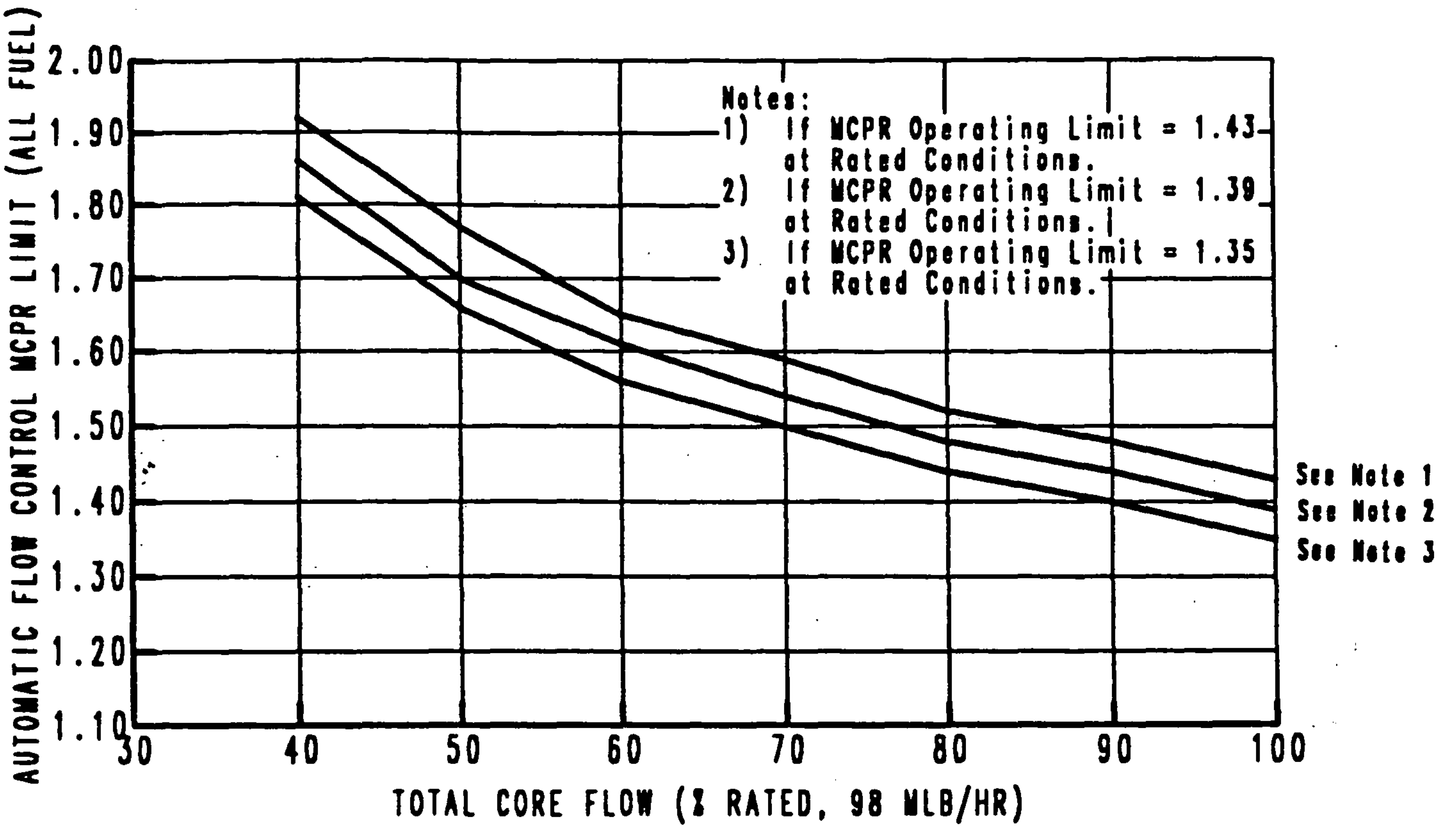
MCPR Operating Limit for Automatic Flow Control



The above curve is based on the following MCPR operating limit summary for Automatic Flow Control and all fuel types:

Total Core Flow (% Rated)	MCPR Operating Limit* 1.35	1.39	1.43
100	1.35	1.39	1.43
90	1.40	1.44	1.48
80	1.44	1.48	1.52
70	1.50	1.54	1.59
60	1.56	1.61	1.65
50	1.66	1.70	1.75
40	1.81	1.86	1.92

* Column headers are MCPR Operating Limits at Rated Flow.

Figure 3.5-2 (Sheet 2 of 2)

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

of 1 containment cooling subsystem leaves one remaining system to perform the containment cooling function. The operable system is demonstrated to be operable each day when the above condition occurs. Based on the facts that when one containment cooling subsystem becomes inoperable only one system remains which is tested daily. A 7-day repair period was specified.

C. High Pressure Coolant Injection - The high pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI ~~or~~ and core spray subsystems can protect the core.

The HPCI meets this requirement without the use of off-site electrical power. For the pipe breaks for which the HPCI is intended to function the core never uncovers and is continuously cooled and thus no clad damage occurs. (Ref. Section 6.2.5.3 SAR). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

D. Automatic Pressure Relief - The relief valves of the automatic pressure relief subsystem are a back-up to the HPCI subsystem. They enable the core spray ~~or~~ and LPCI to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays ~~or~~ and LPCI. The core spray and~~/or~~ LPCI provide sufficient flow of coolant to adequately cool the core.

Insert E

~~Loss of 1 of the relief valves affects the pressure relieving capability and therefore a 7 day repair period is specified. Loss of more than 1 relief valve significantly reduces the pressure relief capability and thus a 24-hour repair period is specified.~~

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

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Analyses have shown that only four of the five valves in the automatic depressurization system are required to operate. Loss of one of the relief valves does not significantly affect the pressure-relieving capability, therefore continued operation is acceptable provided the appropriate MAPLHGR reduction factor is applied to assure compliance with the 2200°F PCT limit. Loss of more than one relief valve significantly reduces the pressure relief capability of the ADS; thus, a 7 day repair period is specified with the HPCI available, and a 24 hour repair period otherwise.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than plus or minus 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 References (1), (2) and (3). Power operation with APLHGRs at or below those shown in Figure 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit.

ANF ~~General Electric~~ has analyzed the effects that Single Loop Operation has on LOCA events (Reference 4). For breaks in the idle loop, the above Dual Loop Operation results are conservative (Reference 1). For breaks in the active loop, the event is more severe primarily due to a more rapid loss of core flow. By ~~decreasing the results of the previous analyses to 70% of the original value, all applicable criteria are met.~~ ~~ENC concurs with GE that the reduction factor is conservatively applicable for cores fueled with 8x8 and 9x9 fuel (Reference 5).~~

applying a multiplicative 0.91 reduction factor to the results of the previous analyses, all applicable criteria are met

The maximum average planar LHGRs for G.E. fuel plotted in Figure 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 Figure 3.5.1 as limits because conformance calculations have not been performed to justify operation at LHGRs in excess of those shown.

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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, Revision 1, April, 1979.

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

(3) XN-NF-85-63 "Dresden Unit 3 LOCA-ECCS Analysis MAPLHGR results for 9x9 fuel", dated September 1985.

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~~(4) NEDO-24807, "Dresden Nuclear Power Station Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2 Single Loop Operation," dated December 1980.~~

~~(5) XN-NF-86-103 "Dresden Unit 2 cycle 11 Reload Analysis" dated September 1986.~~

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(4) ANF-84-111, "LOCA-ECCS Analysis for Dresden Units during Single Loop Operation with ANF Fuel", September 1987.

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3.5 <u>LIMITING CONDITION FOR OPERATION BASES</u> (Cont'd.)

J. <u>Local</u> Steady State <u>LHGR</u>

This specification assures that the maximum linear heat generation rate in any fuel rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. This provides assurance that the fuel end-of-life steady state criteria are met.

(Insert G).

L ~~K~~. <u>Minimum Critical Power Ratio (MCPR)</u>

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 delta 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.

delete

~~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.~~

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K. Local Transient LHGR

This specification provides assurance that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of rated thermal power.

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3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

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 ~~or Sheet 3~~ 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~~, Sheet 3~~ 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 Sheets 2 and 3.~~

Analyses have demonstrated that transient events in Single Loop Operation are bounded by those at rated conditions; however, due to the increase in the MCPR fuel cladding integrity safety limit in Single Loop Operation, an equivalent adder must be uniformly applied to all MCPR LCO to maintain the same margins to the MCPR fuel cladding integrity safety limit.

m ~~K.~~ Flood Protection

Condensate pump room flood protection will assure the availability of the containment cooling service water system (CCSW) during a postulated incident of flooding in the turbine building. The redundant level switches in the condenser pit will preclude any postulated flooding of the turbine building to an elevation above river water level. The level switches provide alarm and circulating water pump trip in the event a water level is detected in the condenser pit.

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4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

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 Steady State LHGR

The LHGR for all 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 permissible control rod pattern below 25% rated thermal power.

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L ~~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 reduced flow correction applied to the LCO provides margin for flow increase from low flows.

M ~~L~~. Flood Protection

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls have been designed to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

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K. Local Transient LHGR

The fuel design limiting ratio for centerline melt (FDLRC) shall be checked daily during reactor operation at greater than or equal to 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution. The FDLRC limit is designed to protect against centerline melting of the fuel during anticipated operational occurrences.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

e. The suction valve in the idle loop shall be closed and electrically isolated except when the idle loop is being prepared for return to service; and

f. If the tripped pump is out of service for more than 24 hours, implement the following additional restrictions:

i. The flow biased RBM Rod Block LSSS shall be reduced by 4.0% (Specification 3.2.C.1);

ii. The flow biased APRM Rod Block LSSS shall be reduced by 3.5% (Specification 2.1.B);

iii. The flow biased APRM scram LSSS shall be reduced by 3.5% (Specification 2.1.A.1);

iv. The MCPR Safety Limit shall be increased by ~~0.03~~ 0.01 (Specification 1.1.A);

v. The MCPR Operating Limit shall be increased by ~~0.03~~ 0.01 (Specification 3.5.~~K~~L.3);

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

vi. The MAPLHGR Operating Limit shall be ~~decreased to 70% of its original value~~ reduced by a multiplicative factor of 0.91 <Insert I> (Specification 3.5.I).

4. Core thermal power shall not exceed 25% of rated without forced recirculation. If core thermal power is greater than 25% of rated without forced recirculation, action shall be initiated within 15 minutes to restore operation to within the prescribed limits and core thermal power shall be returned to within the prescribed limit within two (2) hours.

I. Snubbers (Shock Suppressors)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Snubbers (Shock) Suppressors)

The following surveillance requirements apply to safety related snubbers.

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If, concurrently, one Automatic Pressure Relief Subsystem relief valve is out-of-service, the MAPLHGR Operating Limit shall be reduced by a multiplicative factor of 0.89 for 8x8 fuel and 0.76 for 9x9 fuel.

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3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

In addition, during the start-up of Dresden Unit 2, it was found that a flow mismatch between the two sets of jet pumps caused by a difference in recirculation loops could set up a vibration until a mismatch in speed of 27% occurred. The 10% and 15% speed mismatch restrictions provide additional margin before a pump vibration problem will occur.

Reduced flow MCPR Operating Limits for Automatic Flow Control are not applicable for Single Loop Operation. Therefore, sustained reactor operation under such conditions is not permitted.

Regions I and II of Figure 3.6.2 represent the areas of the power/flow map with the least margin to stable operation. Although calculated decay ratios at the intersection of the natural circulation flow line and the APRM Rod Block line indicate that substantial margin exists to where unstable operation could be expected. Specifications 3.6.H.3.b., 3.6.H.3.c. and 4.6.H.3. provide additional assurance that if unstable operation should occur, it will be detected and corrected in a timely manner.

During the starting sequence of the inoperable recirculation pump, restricting the operable recirculation pump speed below 65% of rated prevents possible damage to the jet pump riser braces due to excessive vibration.

The closure of the suction valve in the idle loop prevents the loss of LPCI through the idle recirculation pump into the downcomer.

Analyses have been performed which support indefinite operation in single loop provided the restrictions discussed in Specification 3.6.H.3.d. are implemented within 24 hours.

The LSSSs are corrected to account for backflow through the idle jet pumps above 40% of rated recirculation pump speed. This assures that the original drive flow biased rod block and scram trip settings are preserved during Single Loop Operation.

The MCPR safety limit has been increased by ~~0.03~~ 0.01 to account for core flow and TIP reading uncertainties which are used in the statistical analysis of the safety limit. In addition, the MCPR Operating Limit has also been increased by ~~0.03~~ 0.01 to maintain the same margin to the safety limit as during Dual Loop Operation.

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3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The decrease of the MAPLHGR Operating Limit to ~~70%~~ 91% of its original value accounts for the more rapid loss of core flow during Single Loop Operation than during Dual Loop Operation.

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Specification 3.6.H.4. increased the margin of safety for thermal-hydraulic stability and for startup of recirculation pumps from natural circulation conditions.

I. Snubbers (Shock Suppressors)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.I.4 prohibits startup with inoperable snubbers.

When a snubber is found inoperable, a review shall be performed to determine the snubber mode of failure. Results of the review shall be used to determine if an engineering evaluation of the safety-related system or component is necessary. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the support component or system.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

All safety related mechanical snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation and attachments to the piping and anchor for indication of damage or impaired operability.

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The more conservative MAPLHGR reduction factors of 0.89 for 8x8 fuel and 0.76 for 9x9 fuel are applied if one relief and one recirculation loop are inoperable at the same time. The small break LOCA is the concern for one relief valve out-of-service; the large break LOCA is the concern for Single Loop Operation. Selecting the more conservative MAPLHGR multipliers will cover both the relief valve out-of-service and Single Loop Operation.

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3.10 <u>LIMITING CONDITIONS FOR OPERATION</u> (Cont'd.)

G. Fuel Storage Reactivity Limit

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

2. Whenever a fuel assembly is stored in the spent fuel storage pool, the peak assembly reactivity in a reactor lattice distribution shall be limited to less than or equal to the following values:

Assembly Type	K_{inf}
GE 7x7	1.26
GE 8x8	1.32
ANF-~~ENC~~ 8x8	1.33
ANF-~~ENC~~ 9x9	1.27

Whenever storing other assembly types or fuel rods in the spent fuel storage pool, their peak reactivity shall be bounded by the most limiting K_{inf} value listed above.

H. Loads Over Spent Fuel Storage Pool

No loads heavier than the weight of a single spent fuel assembly and handling tool shall be carried over fuel stored in the spent fuel storage pool.

4.10 <u>SURVEILLANCE REQUIREMENTS</u> (Cont'd.)

G. Fuel Storage Reactivity Limit

1. Prior to storing Fuel in the new fuel storage facility, an analysis must be performed to demonstrate that the criteria in 3.10.G.1 are satisfied.

2. Prior to storing Fuel in the the spent fuel storage pool, an analysis must be performed to demonstrate that the criteria in 3.10.G.2 are satisfied.

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