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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 95
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated December 10, 1986 as supplemented January 28 and February 5, 1987 complies 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 application, 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 the Technical Specifications as indicated in the attachment to this license amendment and paragraphs 3.B. and 3.M of Provisional Operating License No. DPR-19 are hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 95, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   M. Deleted

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

FOR THE NUCLEAR REGULATORY COMMISSION

Marshall Grotenhuis

, Acting Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 31, 1987

50-237
PROPOSED TECHNICAL SPECIFICATION CHANGES
APPENDIX A

Docket # 50-237
Control # 8704060408
Date 3/31/87 of Document
REGULATORY DOCKET FILE

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ATTACHMENT TO LICENSE AMENDMENT NO. 95

PROVISIONAL OPERATING LICENSE DPR-19

DOCKET NO. 50-237

1. For your convenience revised copies of pages 5 and 6 of Provisional Operating License DPR-19 are attached.

2. Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
iii	iii
vii	vii
viii	viii
1.0-2	1.0-2
1.0-3	1.0-3
1.0-4	1.0-4
1.0-5	1.0-5
1.0-6	1.0-6
1/2.1-1	1/2.1-1
1/2.1-2	1/2.1-2
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B 1/2.1-11	B 1/2.1-11
B 1/2.1-12	B 1/2.1-12
B 1/2.1-13	B 1/2.1-13
B 1/2.1-14	B 1/2.1-14
B 1/2.1-15	B 1/2.1-15*
B 1/2.1-16	B 1/2.1-16*
B 1/2.1-17	B 1/2.1-17
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B 3/4.2-32	B 3/4.2-32
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*Pagination change only

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B 3/4.5-31	3/4.5-31*
B 3/4.5-32	B 3/4.5-32*
B 3/4.5-33	B 3/4.5-33*
B 3/4.5-34	B 3/4.5-34*
B 3/4.5-35	B 3/4.5-35*
B 3/4.5-36	B 3/4.5-36*
B 3/4.5-37	B 3/4.5-37
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B 3/4.5-41	B 3/4.5-41*
B 3/4.5-42	B 3/4.5-42
B 3/4.5-43	B 3/4.5-43*
B 3/4.5-44	B 3/4.5-44*
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B 3/4.6-30	B 3/4.6-30*
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*Pagination change only

Am. 56
2/11/81

I. (3) "Dresden Nuclear Power Station Guard Training and Qualification Plan", submitted by letter dated August 16, 1979, as revised by letter dated August 11, 1980. This Plan shall be full implemented in accordance with 10 CFR 73.55(b)(4), within 60 days of this approval by the Commission. All security personnel shall be qualified within two years of this approval.

J. Systems Integrity

Am. 55
2/06/81

The licensee shall implement a program to reduce leakage from systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspecton requirements, and

2. Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

K. Iodine Monitoring

Am. 55
2/06/81

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel;

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

Am. 53
12/30/80

L. The licensee shall, by January 4, 1981, install a recirculation pump trip, or in the alternative, place and maintain the facility in a cold shutdown or refueling mode of operation.

M. Deleted.

Am. 91
12/12/85

N. Deleted.

DRL
Order
6/10/71

4. This license is effective as of the date of issuance and shall expire December 22, 1972, unless extended for good cause shown, or upon the earlier issurance of a superseding operating license.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by

Peter A. Morris, Director
Division of Reactor Licensing

Attachment:
Appendix A - Technical Specifications
Date of Issuance: December 22, 1969

(Table of Contents, Cont'd.)

List of Tables

List of Figures

1.0 DEFINITIONS (Cont'd.)

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.

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.

1.0 DEFINITIONS (Cont'd.)

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.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

T. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady-state power level of 2527 thermal megawatts.

U. Rated Thermal Power - Rated thermal power means a steady-state power level of 2527 thermal megawatts.

1.0 DEFINITIONS (Cont'd.)

V. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode - In this mode the reactor protection scram trips, initiated by condensor low vacuum and main steamline isolation valve closure, are by-passed when reactor pressure is less than 600 psig; the low pressure main steamline isolation valve closure trip is bypassed, the reactor protection system is energized with IRM neutron-monitoring system trips and control rod withdrawal interlocks in service.

2. Run Mode - In this mode the reactor protection system is energized with APRM protection and RBM interlocks in service.

W. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

X. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

Y. Safety Limit - The safety limits are limits below which the reasonable maintenance of the cladding and primary system are assured. Exceeding such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission (NRC) before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

Z. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All automatic ventilation system isolation valves are operable or are secured in the isolated position.

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).

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".

1.0 DEFINITIONS (Cont'd.)

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.

3960a
3843a

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 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.

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 98×10^6 lb/hr and greater, where:

S – setting in percent of rated thermal power.

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) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

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

FRP = fraction of rated thermal power

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

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.

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. 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) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

S is less than or equal to (.58W_D + 50) [FRP/MFLPD] during Dual Loop Operation or S is less than or equal to (.5 W_D + 46.5) [FRP/ MFLPD] during Single Loop Operation

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.

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.

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 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 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. 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.

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

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

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 during dual loop operation or 116.5 percent during single loop operation, 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.

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.

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.

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 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 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

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

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 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.

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

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.

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.

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

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could lead to a loss of effective core cooling. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Scram - The turbine stop valve 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

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

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 scram set at 10% valve closure, there is no appreciable increase in neutron flux.

I. Turbine Control Valve Fast Closure Scram

The turbine hydraulic control system operates using high pressure oil. There are several points in this oil system where a loss of oil pressure could result in a fast closure of the turbine control valves. This fast closure of the turbine control valves is not protected by the generator load rejection scram since failure of the oil system would not result in the fast closure solenoid valves being actuated. For a turbine control valve fast closure, the core would be protected by the APRM and high reactor pressure scrams. However, to provide the same margins as provided for the generator load rejection scram on fast closure of the turbine control valves, a scram has been added to the reactor protection system which senses failure of control oil pressure to the turbine control system. This is an anticipatory scram and results in reactor shutdown before any significant increase in neutron flux occurs. The transient response is very similar to that resulting from the generator load rejection. The scram setpoint of 900 psig is set high enough to provide the necessary anticipatory function and low enough to minimize the number of spurious scrams. Normal operating pressure for this system is 1250 psig. Finally the control valve will not start to close until the fluid pressure is 600 psig. Therefore, the scram occurs well before valve closure begins.

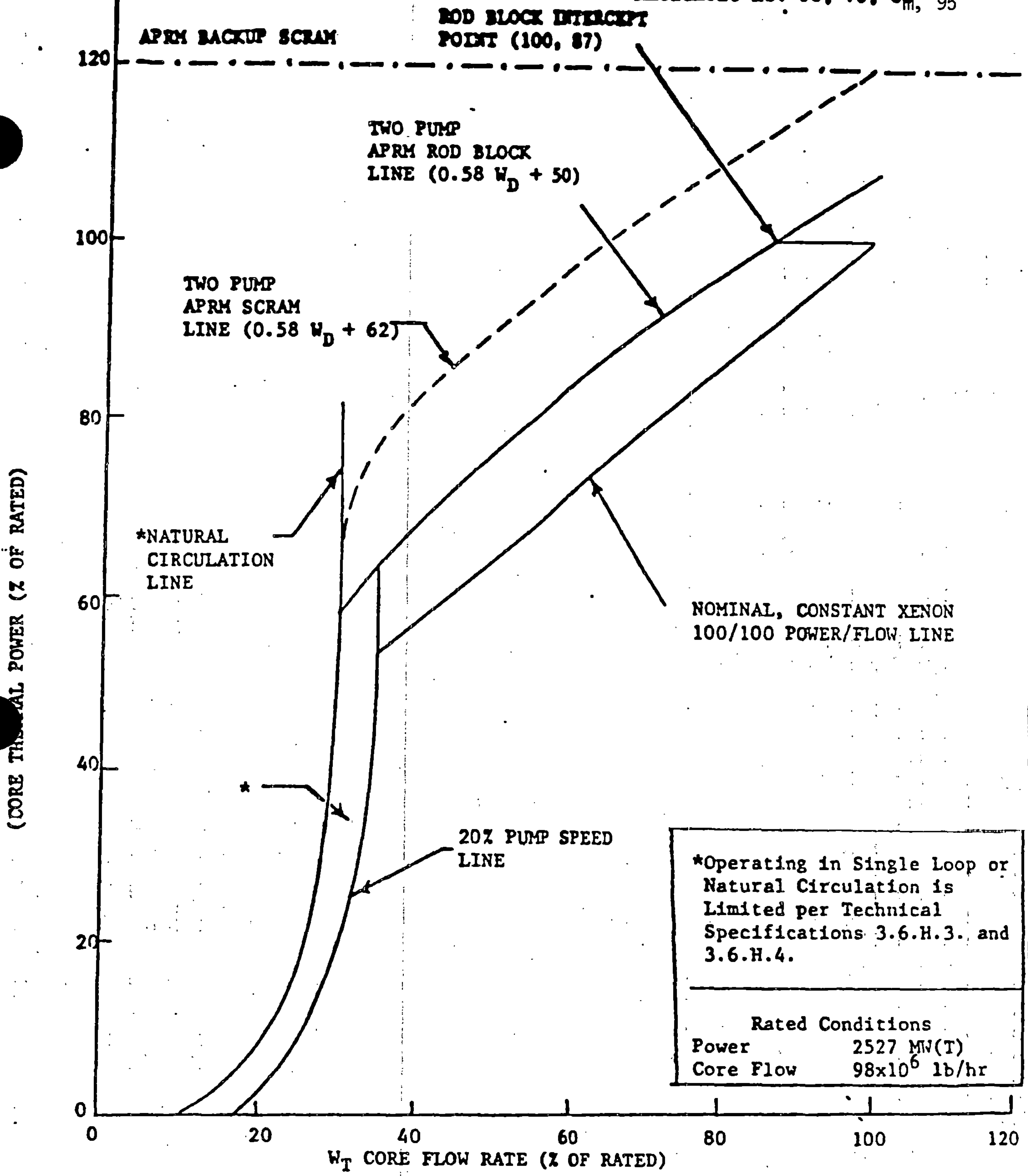

Figure 2.1-3
APRM FLOW BIAS SCRAM RELATIONSHIP TO NORMAL OPERATING CONDITIONS

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:

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:

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.

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).

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 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.

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) (See Note 2)
	Single Loop Operation	Less than or equal to (.58 W_D plus 46.5) (FRP/MFLPD) (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)	
	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.

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.

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

and/or bypass valves to open. With the trip set at 850 psig, inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500 degrees F; thus, there are no fission products available for release other than those in the reactor water. (Ref. Section 11.2.3 SAR)

Two sensors on the isolation condenser supply and return lines are provided to detect the failure of isolation condenser line and actuate isolation action. The sensors on the supply and return sides are arranged in a 1 out of 2 logic and, to meet the single failure criteria, all sensors and instrumentation are required to be operable. The trip settings of 20 psig and 32 inches of water and valve closure time are such as to prevent uncovering the core or exceeding site limits. The sensors will actuate due to high flow in either direction.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI piping. Tripping of this instrumentation results in actuation of HPCI isolation valves, i.e., Group 4 valves. Tripping logic for this function is the same as that for the isolation condenser and thus all sensors are required to be operable to meet the single failure of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

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

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not go below the MCPR fuel cladding integrity safety limit. The trip logic for this function is 1 out of n, e.g., any trip on one of the six APRM's, 8 IRM's, or 4 SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing or calibration. This time period is only approximately 3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal. During Single Loop Operation, the flow biased RBM is reduced by 4 percent to compensate for reverse flow in the idle loop jet pumps.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR, especially during operation at

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

reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence. During Single Loop Operation, the flow biased APRM rod block is reduced by 3.5 percent to compensate for reverse flow in the idle loop jet pumps.

In the refuel and startup/hot standby modes, the APRM rod block function is set at 12% of rated power. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby mode as the APRM flow-biased rod block does in the Run mode, i.e., prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the MCPR fuel cladding integrity safety limit.

Below 30% power, the worst-case withdrawal of a single control rod without rod block action will not violate the MCPR fuel cladding integrity safety limit. Thus, the RBM rod block function is not required below this power level.

The IRM block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity safety limit.

A downscale indication on an APRM or IRM is an indication the instrument has failed or is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and the control rod motion is prevented. The downscale trips are set at 5/125 of full scale.

The rod block which occurs when the IRM detectors are not fully inserted in the core for the refuel and startup/hot standby position of the mode switch has been provided to assure that these detectors are in the core during reactor startup. This, therefore, assures that these instruments are in proper position to provide protection during reactor startup. The IRM's primarily provide protection against local reactivity effects in the source and intermediate neutron range.

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 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 70% of the original value. 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 LHGR

During steady state power operation, 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 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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3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

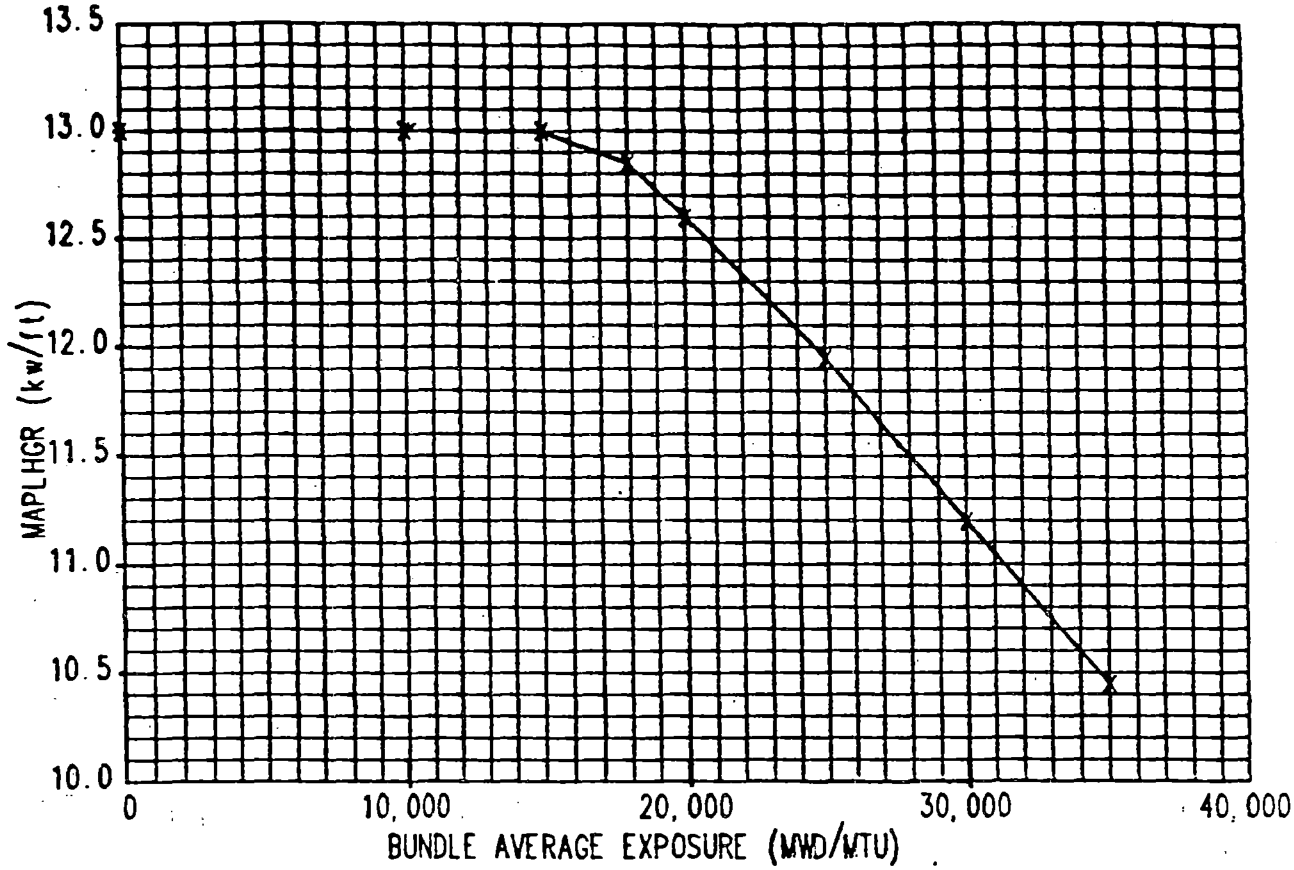
at any axial location shall not exceed its maximum LHGR value shown in Figure 3.5-1A (consists of three curves).

Figure 3.5-1A depicts the LHGR values for Exxon 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, it is determined by normal surveillance that the limiting value for LHGR for any fuel assembly 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 (Cont'd.)

MAPLHGR LIMIT VS. BUNDLE AVERAGE EXPOSURE
ENC 8x8 FUEL



The above graph is based on the following MAPLHGR summary for 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)

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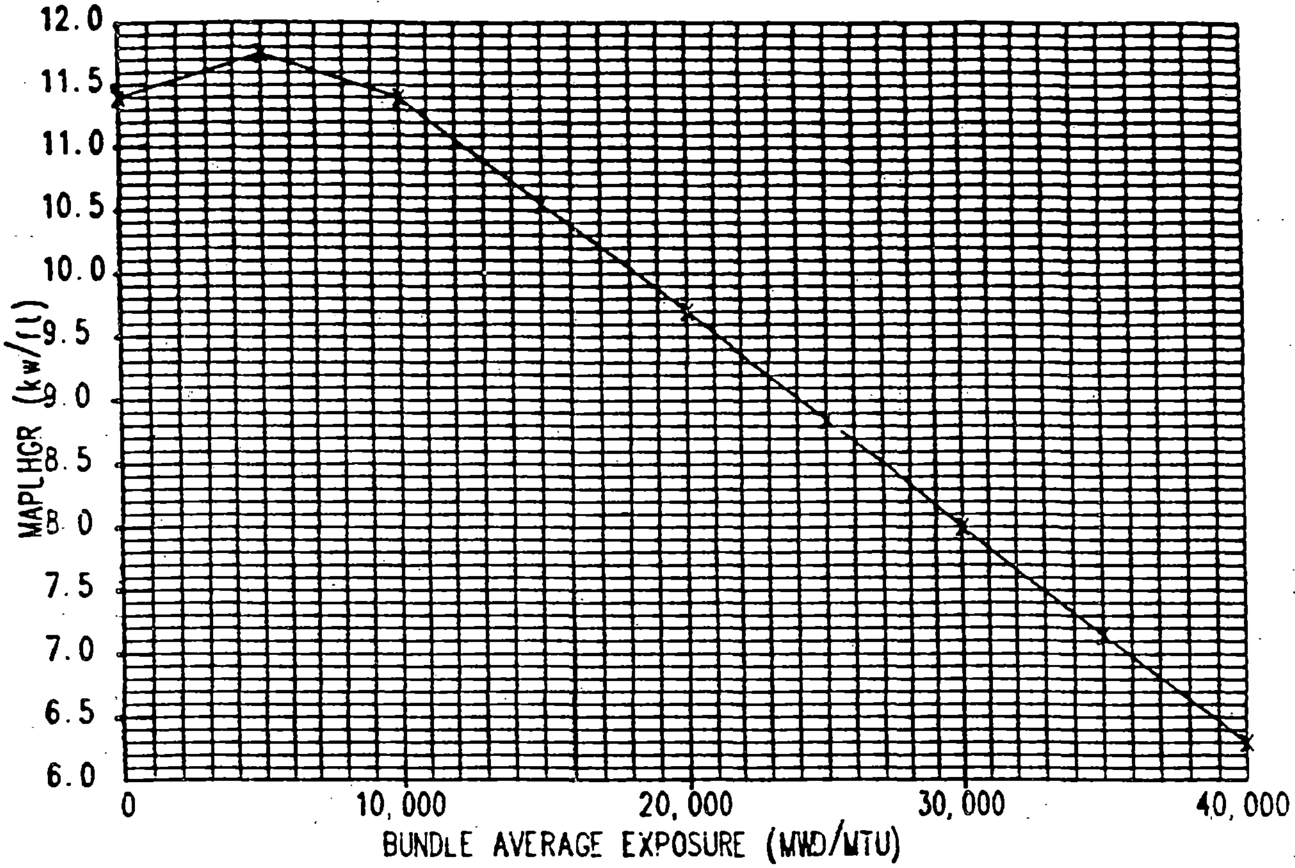
MAPLHGR LIMIT VS. BUNDLE AVERAGE EXPOSURE
ENC 9x9 FUEL

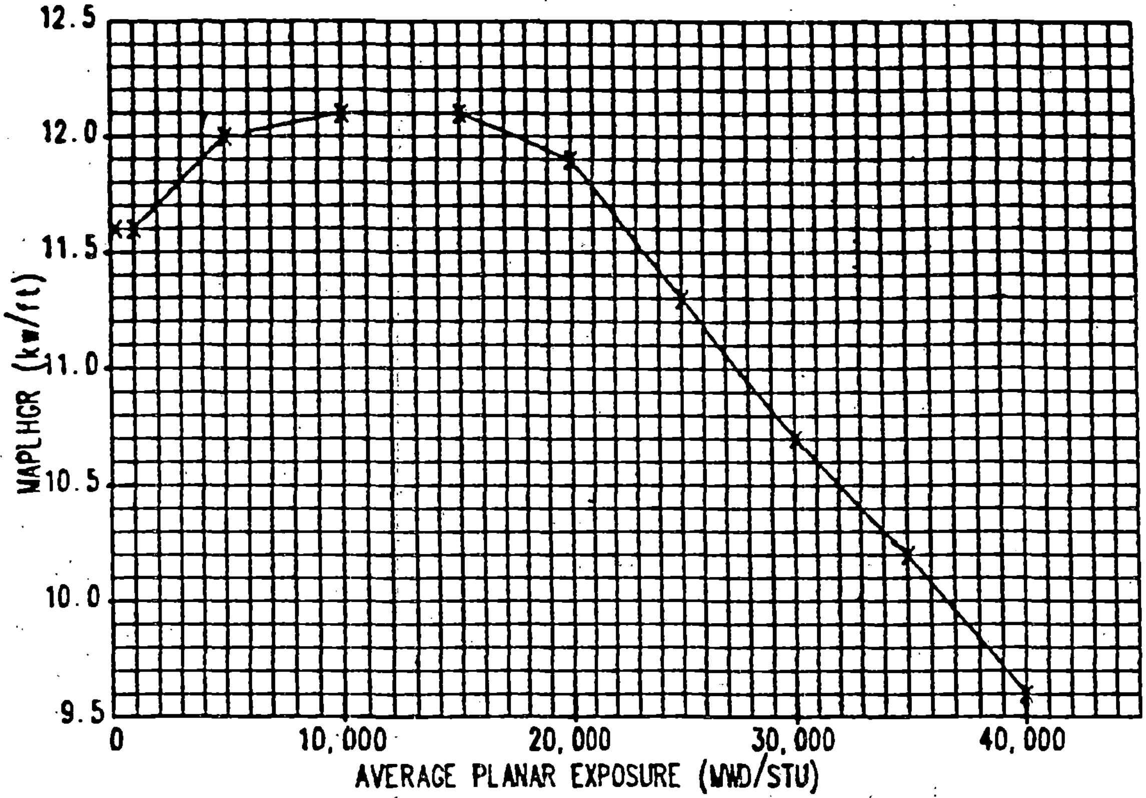


The above graph is based on the following MAPLHGR summary for ENC 9x9 fuel design:

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)

MAPLHGR LIMIT VS. AVERAGE PLANAR EXPOSURE
GE FUEL TYPE P8DRB265L

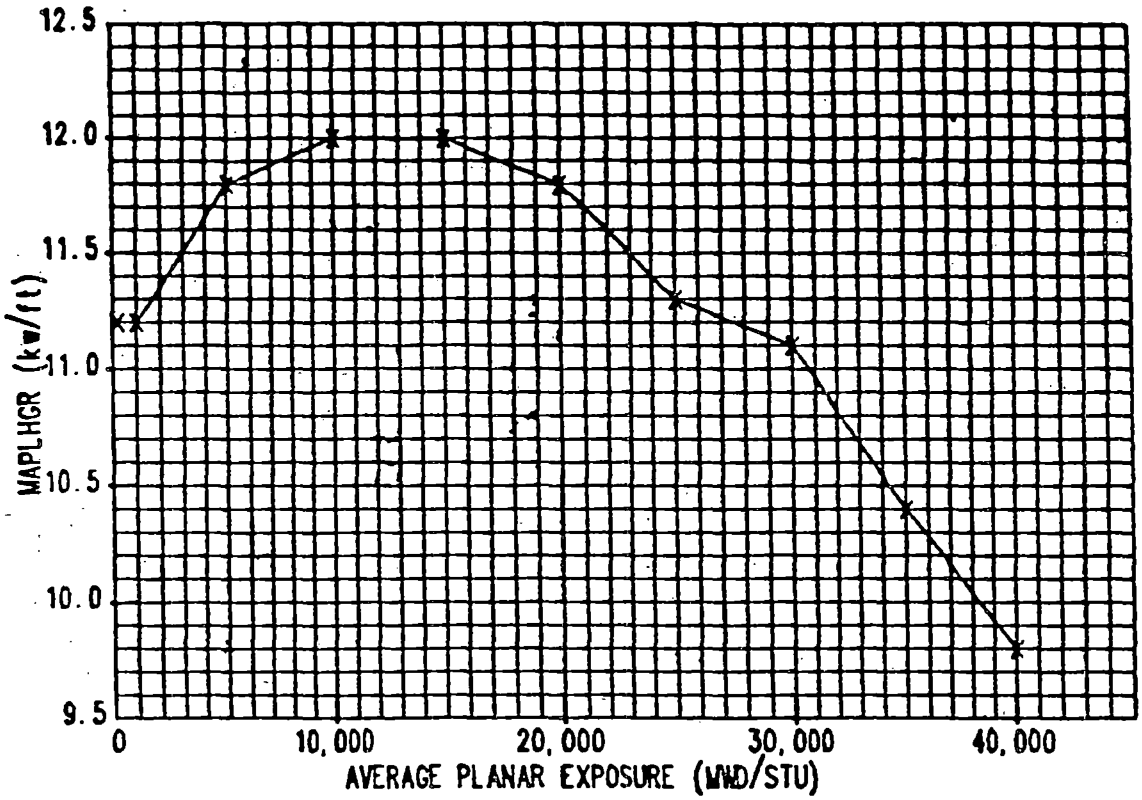


The above graph is based on the following MAPLHGR summary for GE fuel type P8DRB265L.

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)

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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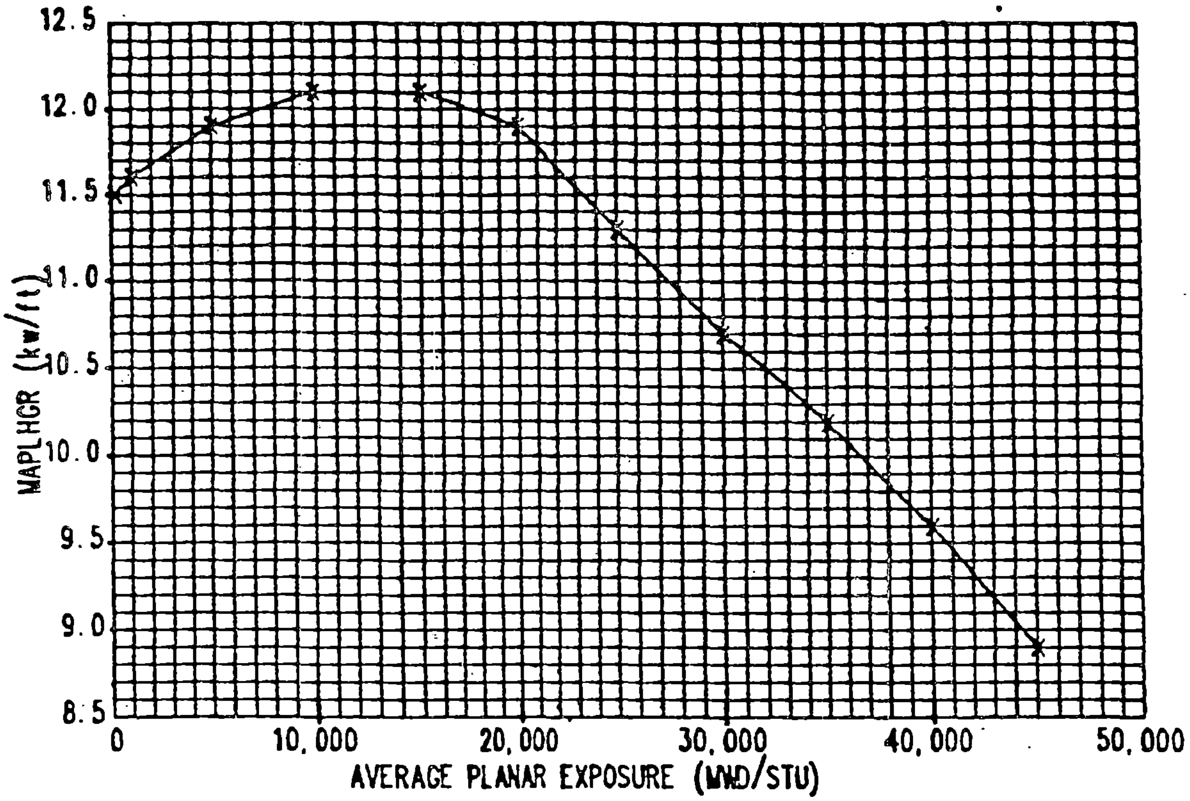
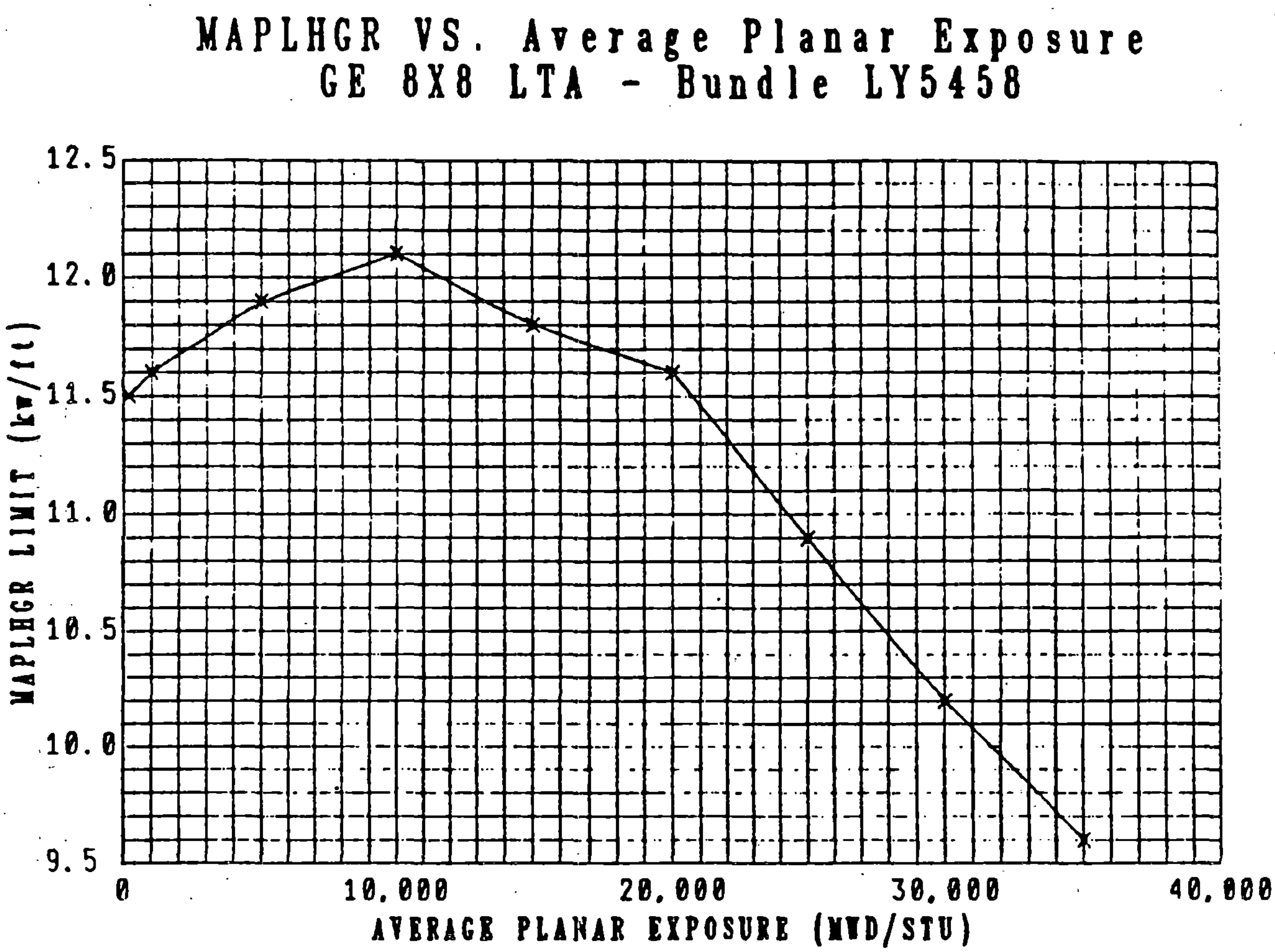
MAPLHGR LIMIT VS. AVERAGE PLANAR EXPOSURE
GE FUEL TYPE P8DRB265H



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

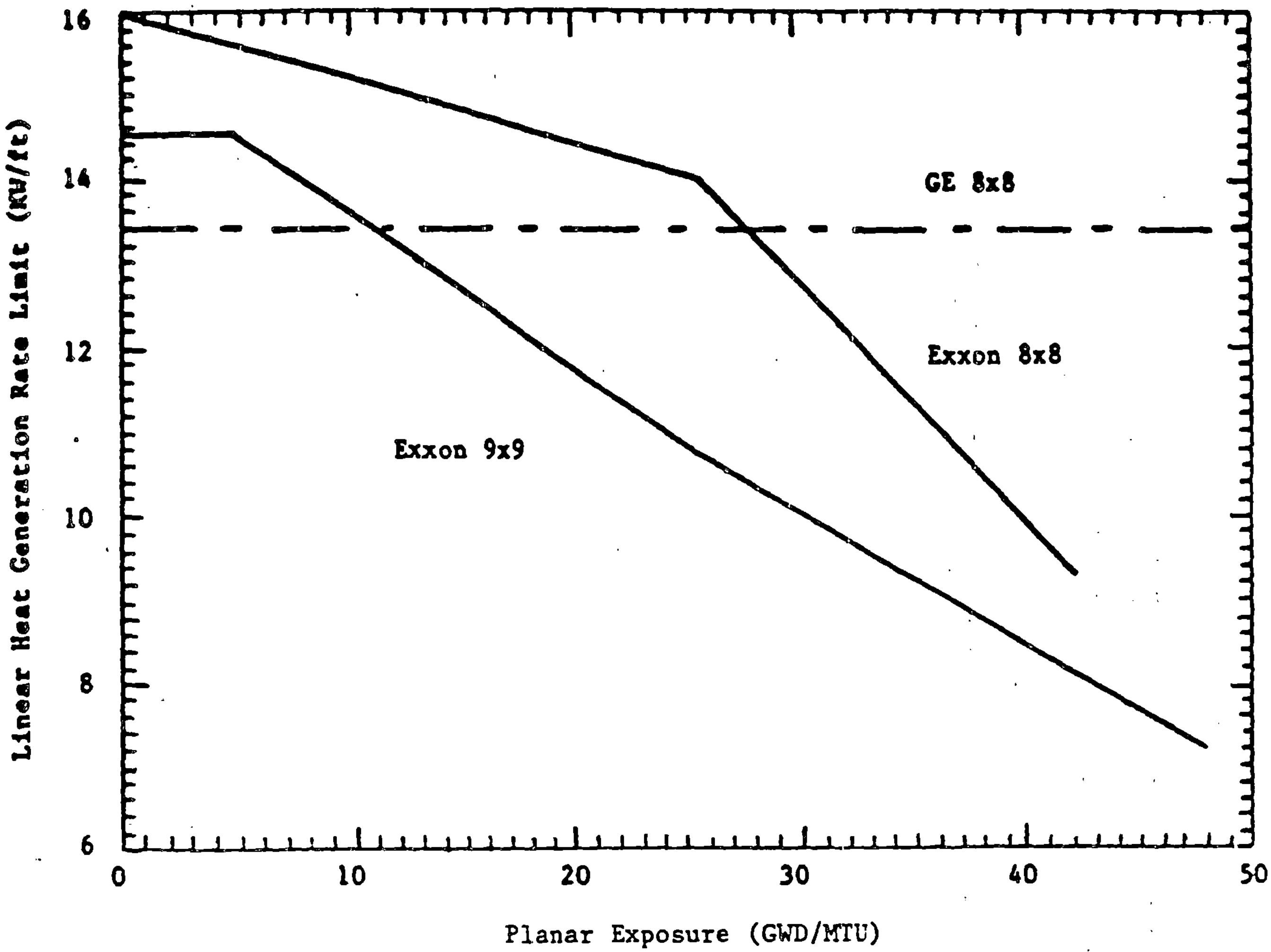
Figure 3.5-1
(Sheet 5 of 6)



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

Figure 3.5-1
(Sheet 6 of 6)



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

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

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

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 sheets 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.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

K. 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.

3.5 LIMITING CONDITION FOR OPERATION (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 SURVEILLANCE REQUIREMENT (Cont'd.)

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

L. Condensate Pump Room Flood Protection

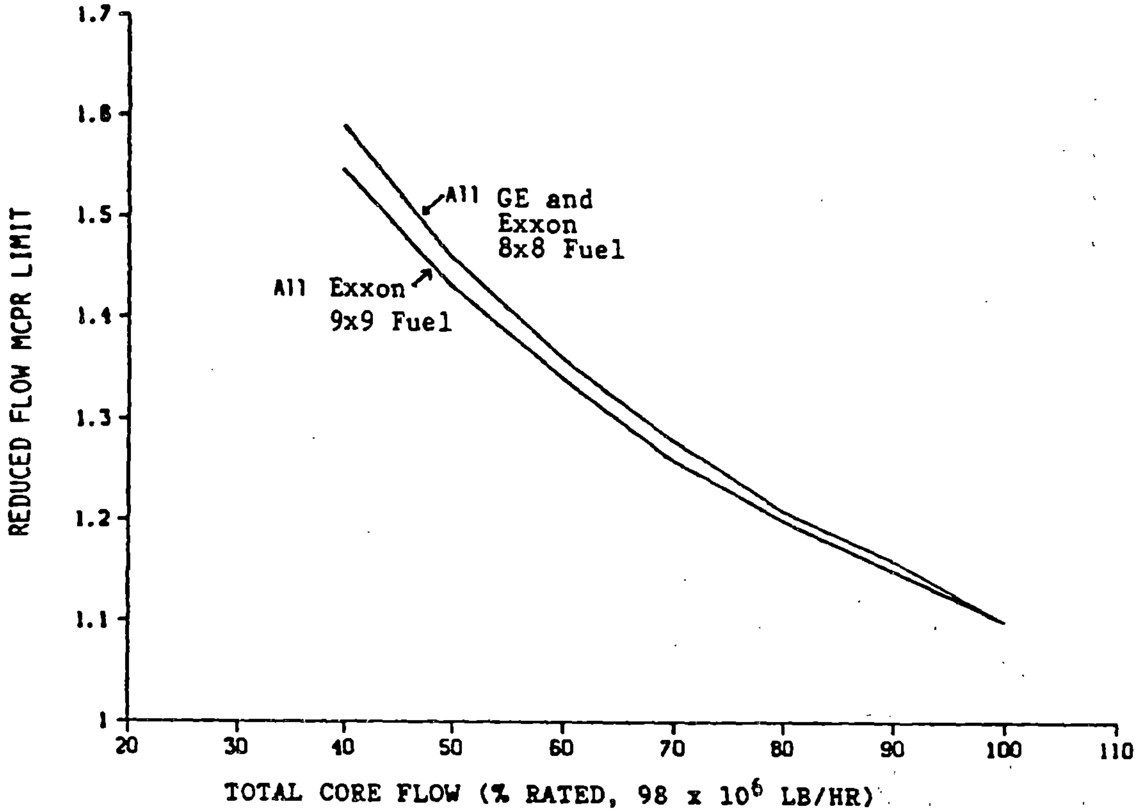
1. The system is installed to prevent or mitigate the consequences of flooding of the condensate pump room shall be operable prior to startup of the reactor.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

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.



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

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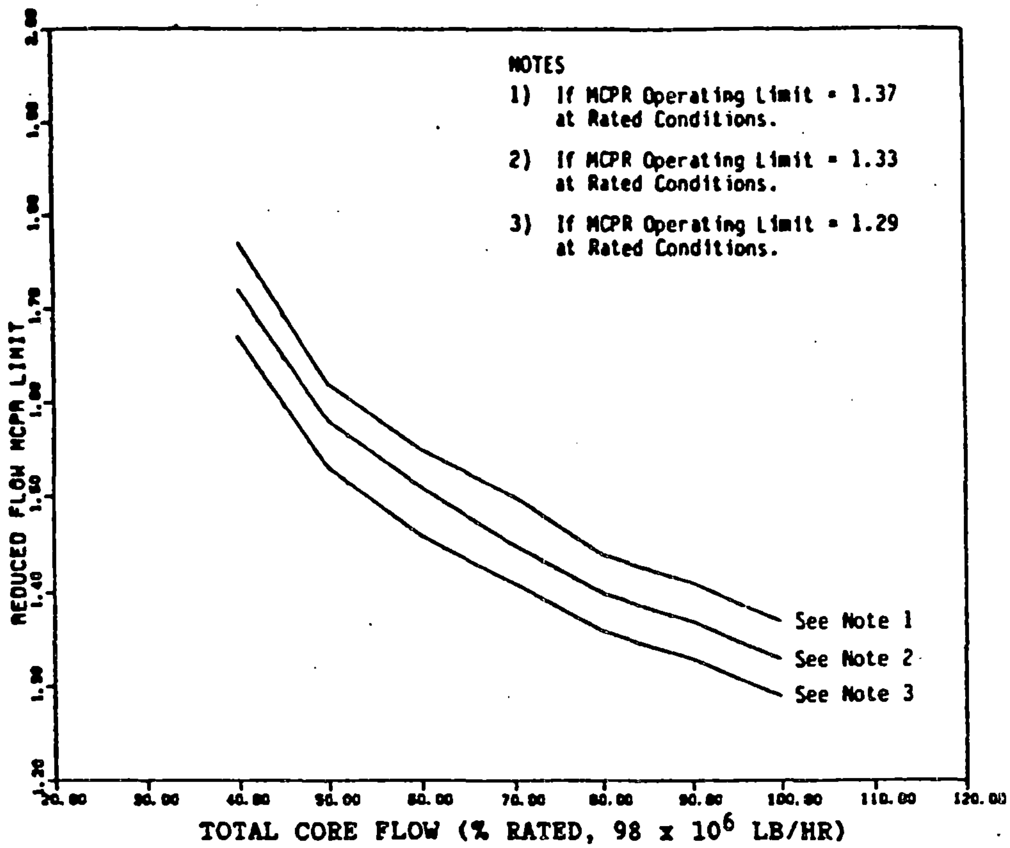

The above GE and Exxon 8x8 curves are based on the following MCPR operating limit summary for Automatic Flow Control:

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

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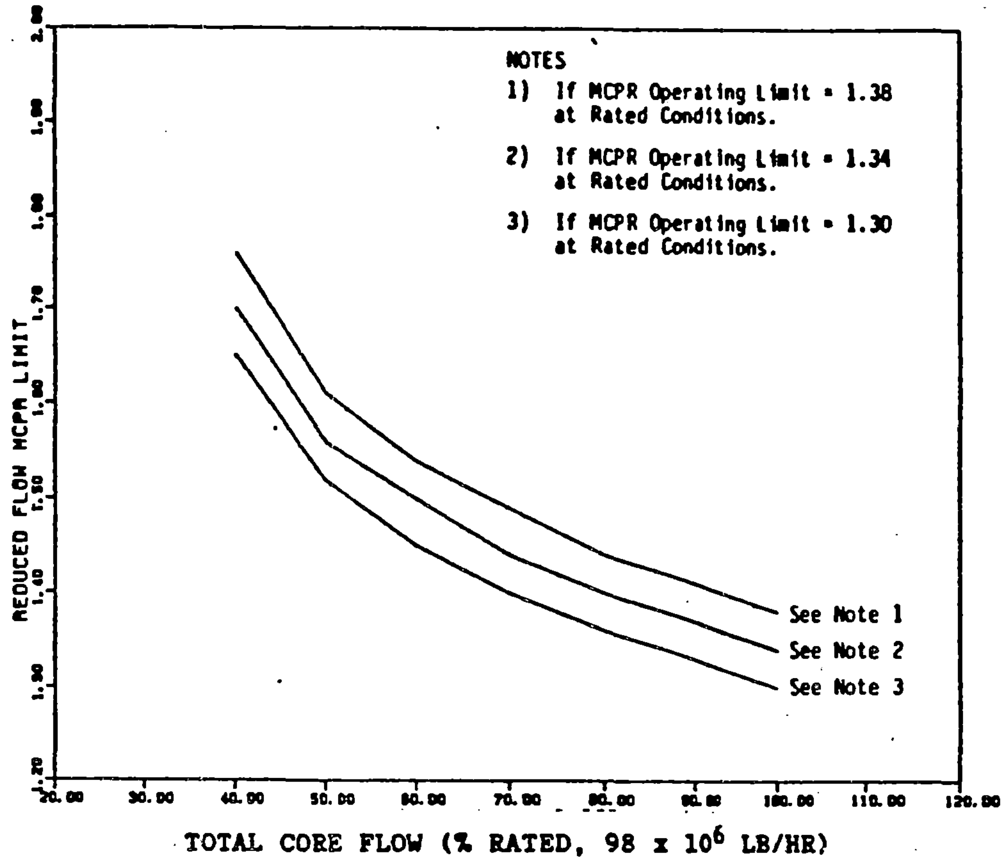

The above Exxon 9x9 curves are based on the following MCPR operating limit summary for Automatic Flow Control:

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

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

b. The CCSW Vault Floor drain shall be checked during each operating cycle by assuring that water can be run through the drain line and actuating the air operated valves by operation of the following sensor:

   i. loss of air

   ii. high level in the condensate pump room (5'0")

c. The condenser pit five foot trip shall have a trip setting of less than or equal to five feet zero inches. The five foot trip circuit for each channel shall be checked once every three months. The 3 and 1 foot alarms shall have a setting of less than or equal to three feet zero inches and less than or equal to 1 foot 0 inches. A logic system functional test, including all alarms, shall be performed during the refueling outage.

3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The condenser pit water level switches shall trip the condenser circulating water pumps and alarm in the control room if water level in the condenser pit exceeds a level of 5 feet above the pit floor. If a failure occurs in one of these trip and alarm circuits, the failed circuit shall be immediately placed in a trip condition and reactor operation shall be permissible for the following seven days unless the circuit is sooner made operable.

3. If Specification 3.5.L.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

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3.5 LIMITING CONDITION 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) and (2) 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

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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, Revisions 1, 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.

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

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 judgement.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the reactor core cooling arise. To assure that the remaining core spray and LPCI subsystems and the diesel generators are available they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves and diesel generators. Based on judgements of the reliability of the remaining systems; i.e. the core spray and LPCI, a 7-day repair period was obtained.

Should the loss of one LPCI pump occur, a nearly full complement of core and containment cooling equipment is available. Three LPCI pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, which will be demonstrated to be operable, a 30-day repair period is justified. If the LPCI subsystem is not available, at least 2 LPCI pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. Containment Cooling Service Water - The containment heat removal portion of the LPCI/containment cooling subsystem is provided to remove heat energy from the containment in the event of a loss of coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and, therefore, is more than ample to provide the required heat removal capability. (Ref. Section 5.2.3.2 SAR).

The containment cooling subsystem consists of two sets of 2 service water pumps, 1 heat exchanger and 2 LPCI pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one containment cooling service water pump does not seriously jeopardize the containment cooling capability as any 2 of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left a 30-day repair period is adequate. Loss

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 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 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 LPCI. The core spray and/or LPCI provide sufficient flow of coolant to adequately cool the core.

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.

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

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1060 psig sustained for 15 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. To be considered operable the shell side of the isolation condenser must contain at least 11,300 gallons of water. Make-up water to the shell side of the isolation condenser is provided by the condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are operable from on-site power. The fire protection system is also available as make-up water. An alternate method of cooling the core upon isolation from the main condenser is by using the relief valves and HPCI subsystem in a feed and bleed manner. Therefore, the high pressure relief function and the HPCI must be available together to cope with an anticipated transient so the LCO for HPCI and relief valves is set upon this function rather than their function as depressurization means for a small pipe break.

F. Emergency Cooling Availability - The purpose of Specification D is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only 2 LPCI pumps would be available. Likewise, if 2 LPCI pumps were out of service and 2 containment service water pumps on the opposite side were also out of service no containment cooling would be available. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

Dresden Units 2 and 3 share certain process systems such as the makeup demineralizers and the radwaste system and also some safety systems such as the standby gas treatment system, batteries, and diesel generators. All of these systems have been sized to perform their intended function considering the simultaneous operation of both units.

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

For the safety related shared features of each plant, the Technical Specifications for that unit contain the operability and surveillance requirements for the shared feature; thus, the level of operability for one unit is maintained independently of the status of the other. For example, the shared diesel (2/3 diesel) would be mentioned in the specifications for both Units 2 and 3 and even if Unit 3 were in the Cold Shutdown Condition and needed no diesel power, readiness of the 2/3 diesel would be required for continuing Unit 2 operation.

G. Specification 3.5.F.4 provides that should this occur, no work will be performed which could preclude adequate emergency cooling capability being available. Work is prohibited unless it is in accordance with specified procedures which limit the period that the control rod drive housing is open and assures that the worst possible loss of coolant resulting from the work will not result in uncovering the reactor core. Thus, this specification assures adequate core cooling. Specification 3.9 must be consulted to determine other requirements for the diesel generator.

Specification 3.5.F.5 provides assurance that an adequate supply of coolant water is immediately available to the low pressure core cooling systems and that the core will remain covered in the event of a loss of coolant accident while the reactor is depressurized with the head removed.

H. Maintenance of Filled Discharge Pipe - If the discharge piping of the core spray, LPCI, and HPCI are not filled, a water hammer can develop in this piping when the pump and/or pumps are started.

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 affects 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

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.

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).

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.

(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.

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

J. Local LHGR

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.

K. Minimum Critical Power Ratio (MCPR)

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.

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.

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.

L. 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.

4.5 SURVEILLANCE REQUIREMENT BASES

(A thru F)

The testing interval for the core and containment cooling systems is based on quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example the core spray final admission valves do not open until reactor pressure has fallen to 350 psig thus during operation even if high drywell pressure were stimulated the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The requirement of 180 psig at 3500 gpm at the containment cooling service water (CCSW) pump discharge provides adequate margin to ensure that the LPCI/CCSW system provides the design

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

bases cooling water flow and maintains 20 psig differential pressure at the containment cooling heat exchanger. This differential pressure precludes reactor coolant from entering the river water side of the containment cooling heat exchangers.

The verification of Main Steam Relief Valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the valve actuation mechanism during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for relief valve opening. This test method may be performed over a wide range of reactor pressure greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

G. Deleted

H. Maintenance of Filled Discharge Pipe

The surveillance requirements to assure that the discharge piping of the core spray, LPCI, and HPCI systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the intervals between the monthly checks the status of the discharge piping is monitored on a continuous basis.

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

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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 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.

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.

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.

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

To test a pipe seal, another test seal is installed in the opposite side of the penetration creating a space between the two seals that can be pressurized. Compressed air is then supplied to a fitting on the test seal and the space inside the sleeve is pressurized to approximately 15 psi. The outer face of the permanent seal is then tested for leaks using a soap bubble solution.

On completion of the test, the test seal is removed for use on other pipes and penetrations of the same size.

In order to test the watertight bulkhead doors, a test frame must be installed around each door. At the time of the test, a reinforced steel box with rubber gasketing is clamped to the wall around the door. The fixture is then pressurized to approximately 15 psig to test for leak tightness.

Floor drainage of each vault is accomplished through a carbon steel pipe which penetrates the vault. When open, this pipe will drain the vault floor to a floor drain sump in the condensate pump room.

Equipment drainage from the vault coolers and the CCSW pump bedplates will also be routed to the vault floor drains. The old equipment drain pipes will be permanently capped to preclude the possibility of back-flooding the vault.

As a means of preventing backflow from outside the vaults in the event of a flood, a check valve and an air operated valve are installed in the 2" vault floor drain line 6'0" above the floor of the condensate pump room.

The check valve is a 2" swing check designed for 125 psig service. The air operated valve is a control valve designed for a 50 psi differential pressure. The control valve will be in the normally open position in the energized condition and will close upon any one of the following:

a. Loss of air or power

b. High level (5'0") in the condensate pump room

Closure of the air operated valve on high water level in the condensate pump room is effected by use of a level switch set at a water level of 5'0". Upon actuation, the switch will close the control valve and alarm in the control room.

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The operator will also be aware of problems in the vaults/condensate pump room if the high level alarm on the equipment drain sump is not terminated in a reasonable amount of time. It must be pointed out that these alarms provide information to the operator but that operator action upon the above alarms is not a necessity for reactor safety since the other provisions provide adequate protection.

A system of level switches has been installed in the condenser pit to indicate and control flooding of the condenser area. The following switches are installed:

	Level	Function
a.	1'0" (1 switch)	Alarm, Panel Hi-Water-Condenser Pit
b.	3'0" (1 switch)	Alarm, Panel High-Circ. Water Condenser Pit
c.	5'0" (2 redundant switch pairs)	Alarm and Circ. Water Pump Trip

Level (a) indicates water in the condenser pit from either the hotwell or the circulating water system. Level (b) is above the hotwell capacity and indicates a probable circulating water failure.

Should the switches at level (a) and (b) fail or the operator fail to trip the circulating water pumps on alarm at level (b), the actuation of either level switch pair at level (c) shall trip the circulating water pumps automatically and alarm in the control room. These redundant level switch pairs at level (c) are designed and installed to IEEE-279, "Criteria for Nuclear Power Plant Protection Systems." As the circulating water pumps are tripped, either manually or automatically, at level (c) of 5'0", the maximum water level reached in the condenser pit due to pumping will be at the 491'0" elevation (10' above condenser pit floor elevation 481'0"; 5' plus an additional 5' attributed to pump coastdown).

In order to prevent overheating of the CCSW pump motors, a vault cooler is supplied for each pump. Each vault cooler is designed to maintain the vault at a maximum 105°F temperature during operation of its respective pump. For example, if CCSW pump 2B-1501 starts, its cooler will also start and compensate

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

for the heat supplied to the vault by the 2B pump motor keeping the vault at less than 105°F.

Each of the coolers is supplied with cooling water from its respective pump's discharge line. After the water has been passed through the cooler, it returns to its respective pump's suction line. In this way, the vault coolers are supplied with cooling water totally inside the vault. The cooling water quantity needed for each cooler is approximately 1% to 5% of the design flow of the pumps so that the recirculation of this small amount of heated water will not affect pump or cooler operation.

Operation of the fans and coolers is required during pump operability testing and thus additional surveillance is not required.

Verification that access doors to each vault are closed, following entrance by personnel, is covered by station operating procedures.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Pressurization Temperature

1. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Curve C of Figure 3.6.1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when vessel temperature is equal to or above that shown in the appropriate curve of Figure 3.6.1. Figure 3.6.1 is effective through 6 effective full power years. At least six months prior to 6 effective full power years new curves will be submitted.

2. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel shell immediately below the vessel flange is greater than or equal to 100°F.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

B. Pressurization Temperature

1. Reactor vessel shell temperature and reactor coolant pressure shall be permanently recorded at 15 minute intervals whenever the shell temperature is below 220°F and the reactor vessel is not vented.

2. When the reactor vessel head bolting studs are tightened or loosened the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

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

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

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

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

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. During Dual Loop Operation, the indicated core flow is the sum of the flow indication from each of the twenty jet pumps. During Single Loop Operation (SLO), the indicated core flow must be conservatively adjusted based on station procedures.

4. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

H. Recirculation Pump Flow Limitations

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

2. If specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

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

H. Recirculation Pump Flow Limitations

Recirculation pumps speed shall be checked daily for mismatch.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. During Single Loop Operation, the following restrictions are required:

   a. Operation with the Master Flow Control in AUTO is not permitted;

   b. Operation in Region I of Figure 3.6-2 is not permitted. Initiate action immediately after entering Region I and be outside of Region I within 2 hours.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.a. Baseline APRM and LPRM* noise levels for SLO shall be acquired prior to entering Region II of Figure 3.6-2 for the first time following each refueling outage.

b. When operating in Region II of Figure 3.6-2 perform APRM and LPRM* surveillances to verify that their noise levels are within three (3) times their established baseline values at the following intervals:

   i. Within 30 minutes of entering Region II;

   ii. At least once per 8 hour shift; and

   iii. Within 30 minutes after the completion of a core thermal power increase of 5% or greater.

* Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center of the core shall be monitored.

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

c. Operation in Region II of Figure 3.6-2 is permitted provided that:

i. Baseline data has been acquired per specification 4.6.H.3.a;

ii. If baseline data has not been acquired per specification 4.6.H.3.a., immediately initiate action to be outside Region II within 2 hours;

iii. Stable reactor operation is verified per specification 4.6.H.3.b.; and

iv. If stable reactor operation cannot be verified per specification 4.6.H.3.b., immediately initiate action to restore stable operation within 2 hours.

d. The operable recirculation pump shall be at a speed less than 65% of rated before starting the inoperable pump;

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3.6 LIMITING CONDITION FOR OPERATION (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 (Specification 1.1.A);

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

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

vi. The MAPLHGR Operating Limit shall be decreased to 70% of its original value (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.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

I. During all modes of operation except cold shutdown and refuel, all safety related snubbers shall be operable except as noted in Specification 3.6.I.2 through 3.6.I.4.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

I. Visual Inspection

An independent visual inspection shall be performed on the safety related hydraulic and mechanical snubbers in accordance with the schedule below.

a. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connection to the piping and anchor to verify snubber operability.

b. All mechanical snubbers shall be visually inspected. This inspection shall consist of, but not necessarily be limited to, inspection of the snubber and attachments to the piping and anchor for indications of damage or impaired operability.

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. From and after the time a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

No. of Snubbers Found Inoperable During Inspection Interval	Next Required Inspection Interval
0	18 months plus or minus 25%
1	12 months plus or minus 25%
2	6 months plus or minus 25%
3,4	124 days plus or minus 25%
5,6,7	62 days plus or minus 25%
8 or more	31 days plus or minus 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible," based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

2. Functional Testing

a. Once each refueling cycle, a representative sample of approximately 10% of the hydraulic snubbers shall be functionally tested for operability, including:

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

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

(i) Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

(ii) Snubber bleed, or release rate, where required, is within the specified range in compression or tension.

For each unit and subsequent unit found inoperable, an additional 10% of the hydraulic snubbers shall be tested until no more failures are found or all units have been tested.

b. Once each refueling cycle, a representative sample of approximately 10% of the mechanical snubbers shall be functionally tested for operability. The test shall consist of two parts:

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

(i) Verification that the force that initiates free movement of the snubber in either tension or compression is less than the specified maximum breakaway friction force.

(ii) Verify that the activation (restraining action) is achieved within the specified range of acceleration or velocity, as applicable based on snubber design in both tension and compression.

For each unit and subsequent unit found inoperable, an additional 10% of the mechanical snubbers shall be so tested until no more failures are found or all units have been tested.

c. In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the resampling.

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown or refuel condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the cold shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3. When a snubber is deemed inoperable, a review of all pertinent facts shall be conducted to determine the snubber mode of failure and to decide if an engineering evaluation should be performed on the supported system or components. If said evaluation is deemed necessary, it will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

4. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and, if determined to be a generic deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

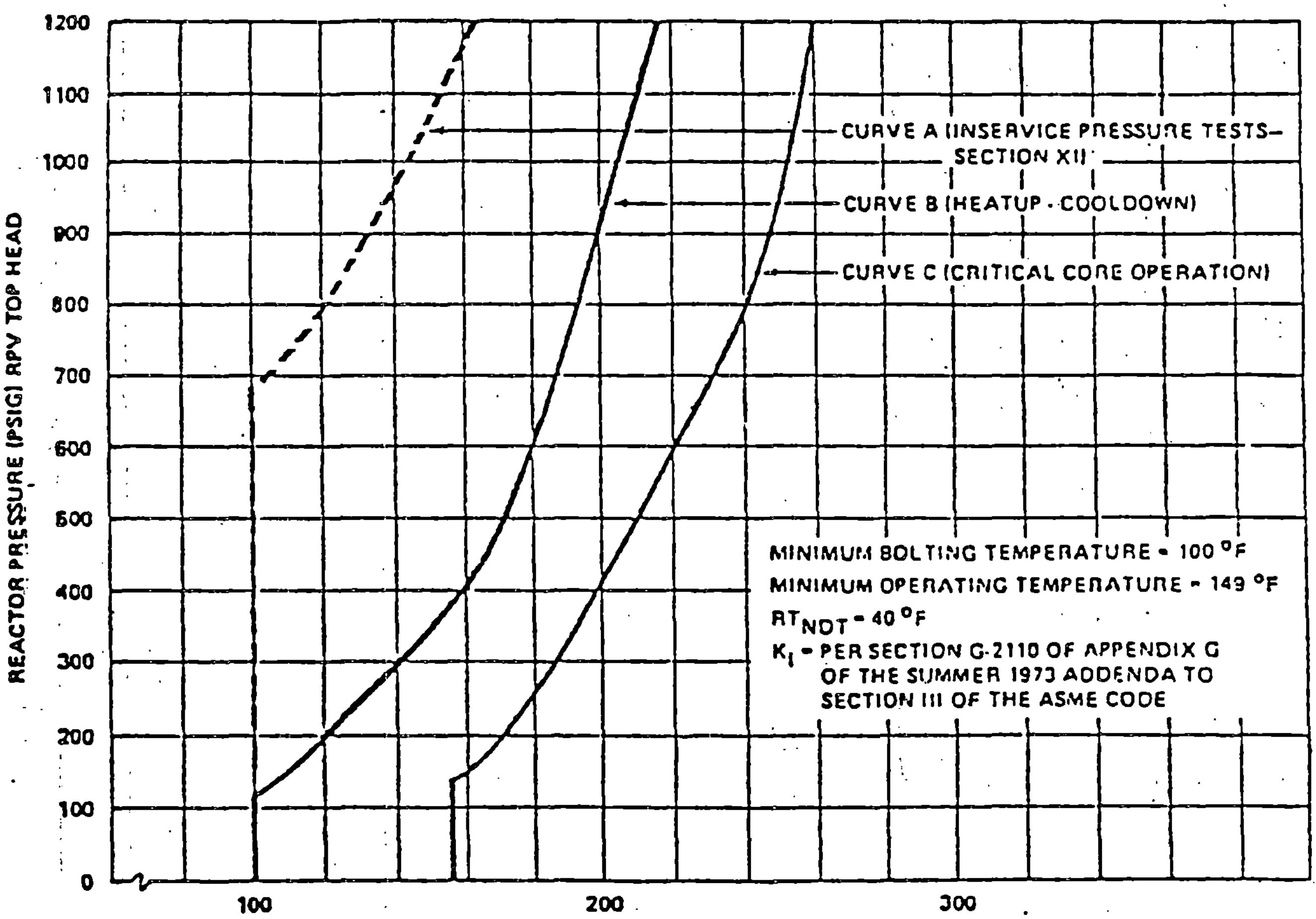
3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

5. Snubbers may be added or removed from safety related systems without prior license amendment.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

5. Snubber service life monitoring shall be followed by existing station record systems, including the central filing system, maintenance files, safety related work packages, and snubber inspection records. The above record retention methods shall be used to prevent the hydraulic snubbers from exceeding a service life of 10 years and the mechanical snubbers from exceeding a service life of 40 years (lifetime of the plant).

MINIMUM TEMPERATURE REQUIREMENTS
PER APPENDIX G OF 10 CFR 50



AVERAGE BULK MODERATOR
TEMPERATURE (°F)

FIGURE 3.6.1.

Minimum Temperature Requirements per Appendix G of 10CFR50

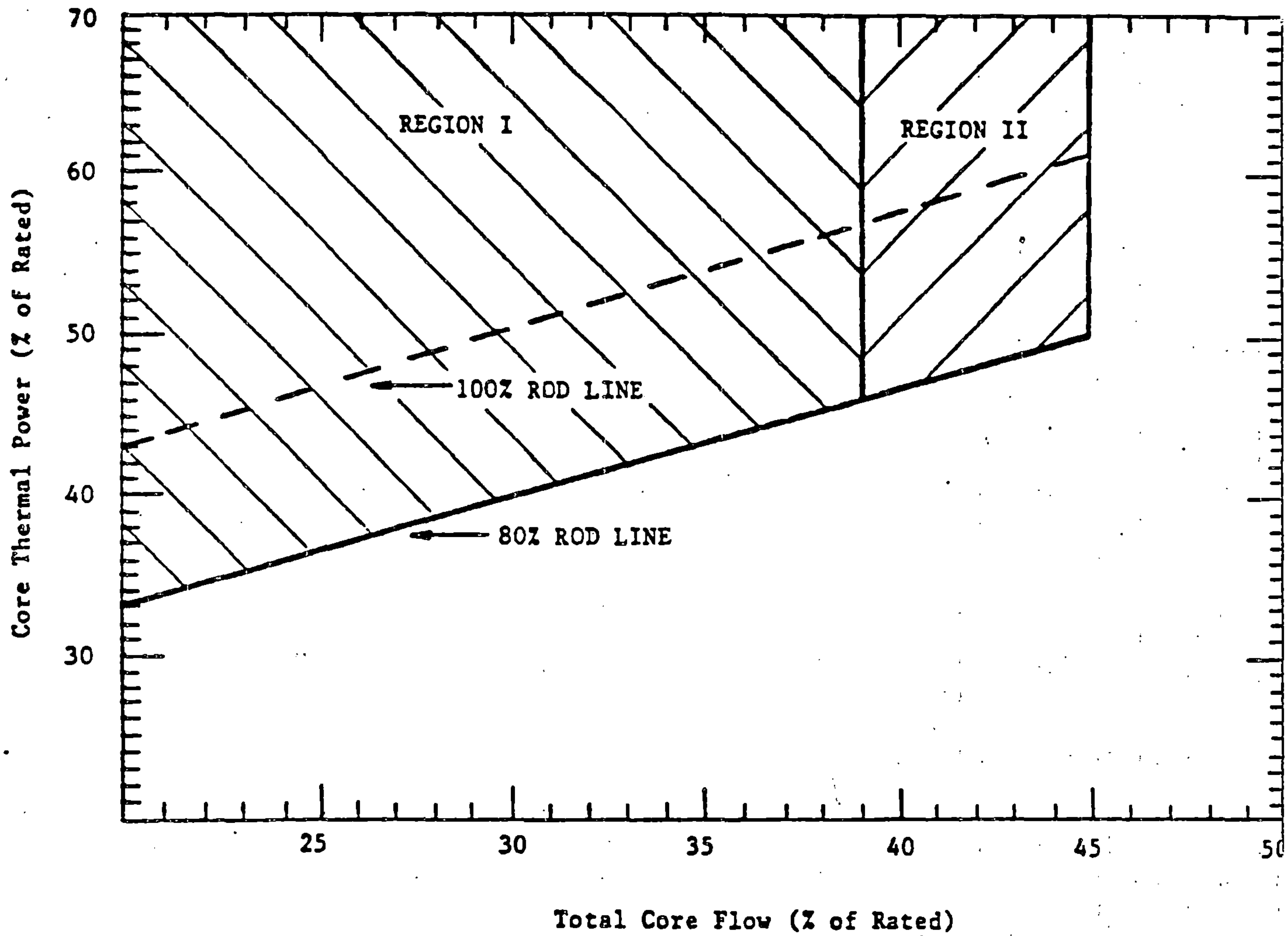


FIGURE 3.6.2

THERMAL POWER VS CORE FLOW LIMITS FOR THERMAL HYDRAULIC STABILITY SURVEILLANCE IN SINGLE LOOP OPERATION

3.6 LIMITING CONDITION FOR OPERATION BASES

A. Thermal Limitations - The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 550°F. Because of the slow temperature-time response of the massive flanges relative to the adjacent head and shell sections, calculated temperatures obtained were 500°F (shell) and 360°F (flange) (140°F differential). Both axial and radial thermal stresses were considered to act concurrently with full primary loadings. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

The flange metal temperature differential of 140°F occurred as a result of sluggish temperature response and the fact that the heating rate continued over a 450°F coolant temperature range.

The uncontrolled cooldown rate of 240°F was based on the maximum expected transient over the lifetime of the reactor vessel. This maximum expected transient is the injection of cold water into the vessel by the high pressure coolant injection subsystem. This transient was considered in the design of the pressure vessel and five such cycles were considered in the design. Detailed stress analyses were conducted to assure that the injection of cold water into the vessel by the HPCI would not exceed ASME stress code limitations.

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

B. Pressurization Temperature - The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected. These restrictions on inservice hydrostatic testing, on heatup and cooldown, and on critical core operation shown in Figure 3.6.1, were established to be in conformance with Appendix G to 10 CFR 50.

In evaluating the adequacy of ferritic steels Sa302B it is necessary that the following be established:

a) The reference nil-ductility temperature (RT_{NDT}) for all vessel and adjoining materials,

b) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev), and

c) the fluence at the location of a postulated flow.

The initial RT_{NDT} of the main closure flange, the shell and head materials connecting to these flanges, and connecting welds is 10°F. However, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. (Reference Appendix F to the FSAR) The closure flanges and connecting shell materials are not subject to any appreciable neutron radiation exposure, nor are the vertical electroslag seams. The flange area is moderately stressed by tensioning the head bolts. Therefore, as is indicated in curves (a) and (b) of Figure 3.6.1, the minimum temperature of the vessel shell immediately below the vessel flange is established as 100°F below a pressure of 400 psig. (40°F + 60°F, where 40°F is the RT_{NDT} of the electroslag weld and 60°F is a conservatism required by the ASME Code). Above approximately 400 psig pressure, the stresses associated with pressurization are more limiting than the bolting stresses, a fact that is reflected in the non-linear portion of curves (a) and (b). Curve (c), which defines the temperature limitations for critical core operation, was established per Section IV 2.c. of Appendix G of 10CFR50. Each of the curves, (a), (b) and (c) define temperature limitations for unirradiated ferric steels. Provision has been made for the modification of these curves to account for the change in RT_{NDT} as a result of neutron embrittlement.

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

The withdrawal schedule in Table 4.6.2 is based on the three capsule surveillance program as defined in Section 11.C.3.a of 10 CFR 50 Appendix H. The accelerated capsule (Near Core Top Guide) is not required by Appendix H but will be tested to provide additional information on the vessel material.

This surveillance program conforms to ASTM E 185-73 "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" with one exception. The base metal specimens of the vessel were made with their longitudinal axes parallel to the principal rolling direction of the vessel plate.

C. Coolant Chemistry - A radioactivity concentration limit of 20 Micro-Ci/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Specification 3.8.C.1 or there is a failure or a prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture, outside the drywell, the resultant radiological dose at the site boundary would be about 10 rem to the thyroid. This dose was calculated on the basis of a total iodine activity limit of 20 Micro-Ci/ml, meteorology corresponding to Type F conditions with a one meter per second wind speed, and a valve closure time of five seconds. If the valve closed in ten seconds, then the resultant dose would increase to about 25 rem.

The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hours. In addition, the trend of the stack off-gas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam lines.

Materials in the primary system are primarily 304 stainless steel and the Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of

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

the stainless steel. The attached graph, Figure 4.6.2, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup and hot standby. During these periods with steaming rates less than 100,000 pounds per hour, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Figure 4.6.2 are not exceeded. At steaming rates of at least 100,000 pounds per hour, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This would not necessarily be the case. Conductivity could be high due to the presence of a neutral salt; e.g., Na_2SO_4, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the off-standard condition include, operation of the reactor 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.

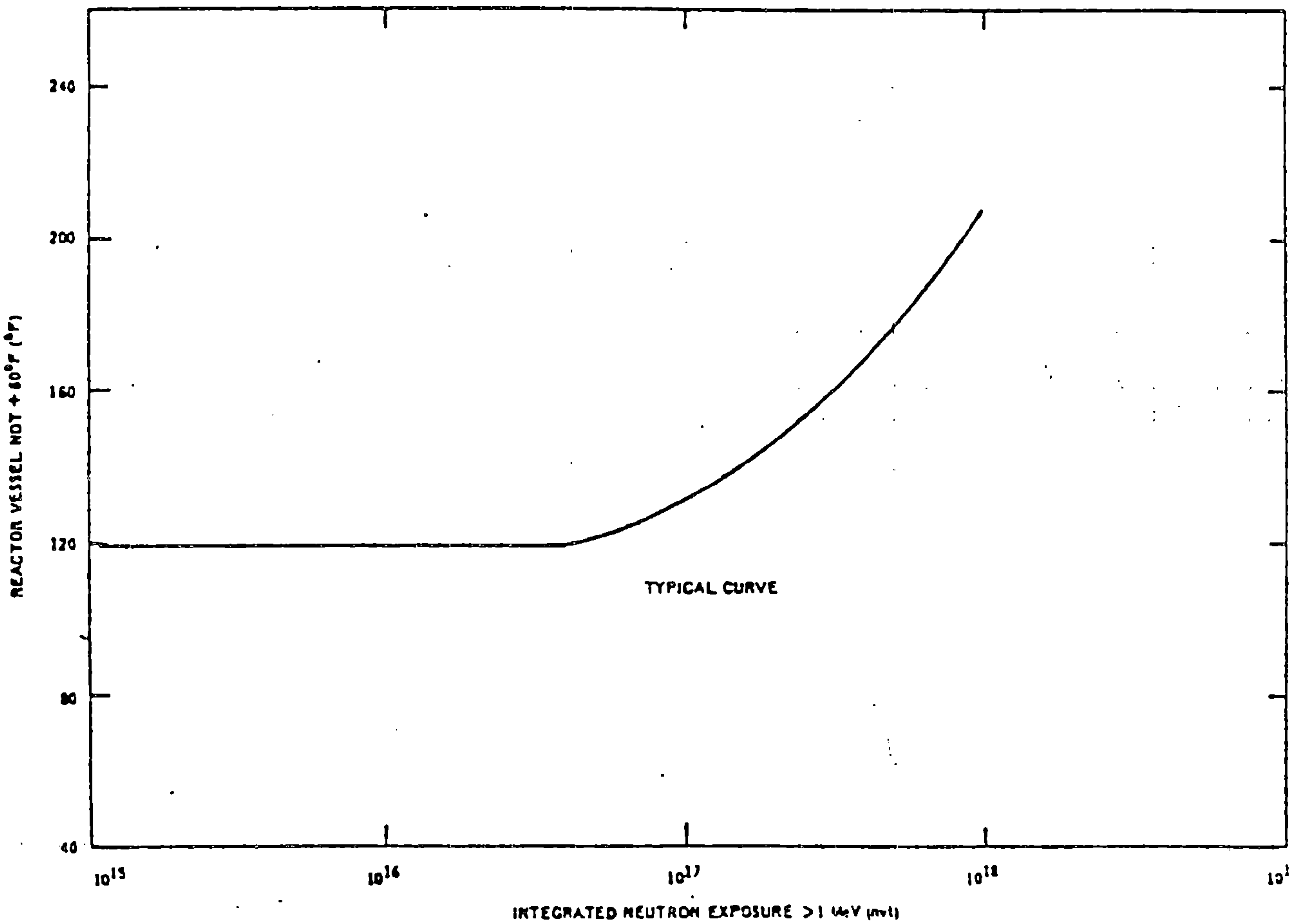


Figure 4.6.1

MINIMUM REACTOR PRESSURIZATION TEMPERATURE

TABLE 4.6.2

NEUTRON FLUX AND SAMPLES WITHDRAWAL
SCHEDULE FOR DRESDEN UNIT 2

Withdrawal* Year	Part No.	Location	Comments
1977	6	Near Core Top/ Guide - 180°	Accelerated Sample
1980	8	Wall - 215°	
2000	7	Wall - 95°	
	9	Wall - 245°	Standby
	10	Wall - 275°	Standby

*Allowances should be made to withdrawal year due to unscheduled shutdowns and updated fuel exposure data.

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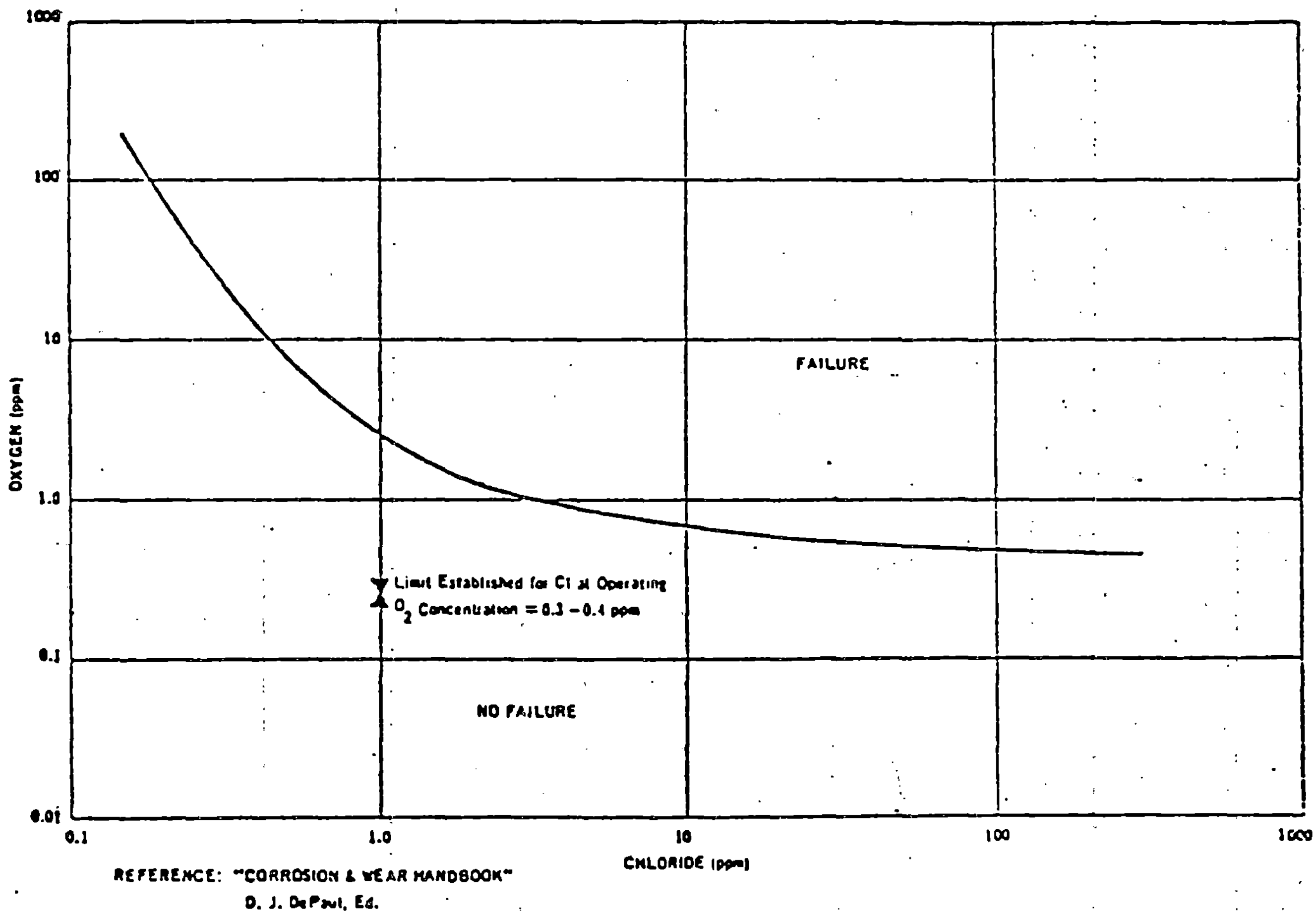

Figure 4.6.2

CHLORIDE STRESS CORROSION TEST RESULTS AT 500°F

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

During start-up periods, which are in the category of less than 100,000 pounds per hour, conductivity may exceed 2 micro-mho/cm because of the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 micro-mho (other than short term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of 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 background 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

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

leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shut down 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.

The performance of reactor coolant leakage detection system will be evaluated during the first five years of station operation and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detection system is capable of detecting the order of 3000 lb/hr.

The system performance will be evaluated during the first five years of plant operation and the conclusions of the evaluation will be reported to the NRC.

E. Safety and Relief Valves - The frequency and testing requirements for the safety and relief valves are specified in the Inservice Testing Program which is based on Section XI of the ASME Boiler and Pressure Vessel Code. Adherence to these code requirements provides adequate assurance as to the proper operational readiness of these valves. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as plus or minus 1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher than the reactor coolant pressure safety limit of 1375 psig is not exceeded. The safety valves are required to be operable above the design pressure (90 psig) at which the core spray subsystems are not designed to deliver full flow.

F. Structural integrity - A pre-service inspection of the components in the primary coolant pressure boundary will be conducted after site erection to assure the system is free of gross defects and as a reference base for later inspections. Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout life.

Inservice Inspections of ASME Code Class 1, 2 and 3 components will be performed in accordance with the applicable version of Section XI of the ASME Boiler and Pressure Vessel Code.

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

Relief from any of the above requirements must be provided in writing by the Commission. The Inservice Inspection program and the written relief do not form a part of these Technical Specifications.

These studies show that it requires thousands of stress cycles at stresses beyond any expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

After five years of operation, a program for in-service inspection of piping and components within the primary pressure boundary which are outside the downstream containment isolation valve shall be submitted to the NRC.

G. Jet Pumps - Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser increases the cross sectional flow area for blowdown following the postulated design basis double-ended recirculation line break. Therefore, if a failure occurs, repairs must be made to assure the validity of the calculated consequences.

The following factors form the basis for the surveillance requirements:

A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

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

The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Section 4.6.G.1 and 2.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive or broken jet pumps. This bypass flow is reverse with respect to normal jet pump flow. During Single Loop Operation, reverse flow through the idle jet pumps begins when the active loop recirculation pump speed is above 20 to 40% of rated. The indicated total core flow is a summation of the flow indications for the twenty individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow. Thus the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operatng map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle riser system failure.

H. Recirculation Pump Flow Limitations

The LPCI loop selection logic has been described in the Dresden Nuclear Power Station Units 2 and 3 FSAR, Amendments 7 and 8. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions, the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. Below 80% power, the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

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 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 to maintain the same margin to the safety limit as during Dual Loop Operation.

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

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

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.

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

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To further increase the assurance of snubber reliability, functional tests will be performed once each refueling cycle. A representative sample of 10% of the safety-related snubbers will be functionally tested. Observed failures on these samples will require testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as different entities for the above surveillance programs.

Hydraulic snubber testing will include stroking of the snubbers to verify piston movement, lock-up, and bleed. Functional testing of the mechanical snubbers will consist of verification that the force that initiates free movement of the snubber in either tension or compression is less than the maximum breakaway friction force and verification that the activation (restraining action) is achieved within the specified range of acceleration or velocity, as applicable based on snubber design, in both tension and compression.

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

When the cause of rejection of the snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

Monitoring of snubber service life shall consist of the existing station record systems, including the central filing system, maintenance files, safety-related work packages, and snubber inspection records. The record retention programs employed at the station shall allow station personnel to maintain snubber integrity. The service life for hydraulic snubbers is 10 years. The hydraulic snubbers existing locations do not impose undue safety implications on the piping and components because they are not exposed to excesses in environmental conditions. The service life for mechanical snubbers is 40 years, lifetime of the plant. The mechanical snubbers are installed in areas of harsh environmental conditions because of their dependability over hydraulic snubbers in these areas. All snubber installations have been thoroughly engineered providing the necessary safety requirements. Evaluations of all snubber locations and environmental conditions justify the above conservative snubber service lives.

4.6 SURVEILLANCE REQUIREMENT BASES

None