Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

May 27, 1974

Regulatory Docket File

Mr. J. F. O'Leary, Director
Directorate of Licensing
Office of Regulation
U.S. Atomic Energy Commission
Washington, D.C. 20545

Subject: Dresden Station Unit 3 - Proposed Change to Facility Operating License DPR-25, AEC Dkt 50-249

Dear Mr. O'Leary:

Pursuant to Section 50.59 of 10 CFR 50 and Paragraph 3.B of Facility Operating License DPR-25, Commonwealth Edison Company hereby submits a proposed change to Appendix A of DPR-25. The purpose of this change is to modify the limiting safety settings for the APRM flux scram and the APRM rod block to appropriate values for both 7 x 7 and 8 x 8 matrix fuel. The proposed changes are indicated on attached revised Pages 2, 4, 5, 5A, 6, 7, 9, 11, 14, 16, 16A, 17, 22 and 34.

These changes have been reviewed and on the basis that the limits maintain the previous safety margins applied to 7 x 7 fuel and establish similar margins for the 8 x 8 fuel, it has been determined that no new or unreviewed safety consideration is involved with this proposed change. This conclusion is supported by the proposed change to Figure 2.1-2 (Page 9) showing that the limiting safety system settings are based on the design peak linear heat generation rate of both 7 x 7 and 8 x 8 fuel for the 100% power conditions.

This proposed change has received both On- and Off-Site review and approval.

Three (3) signed originals and 37 copies of this proposed change are submitted for your review.

Very truly yours,

J. S. Abel
Nuclear Licensing Administrator
Boiling Water Reactors

SUBSCRIBED and SWORN to before me this 27th day of May, 1974.

Brenda Pammer
Notary Public

5021

and abnormal situations can be safely controlled.

I. 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.

J. Limiting Total Peaking Factor - The Limiting Total Peaking Factor (LTPF) is the lowest Total Peaking Factor which limits a fuel type to a Linear Heat Generation Rate (LHGR) corresponding to the operating limit at 100% power.

K. Logic System Functional 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.

L. Minimum Critical Heat Flux Ratio (MCHFR) — The lowest in-core ratio of critical heat flux (that heat flux which results in transition boiling) to the actual heat flux.

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

N. Operable — A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

O. Operating — Operating means that a system or component is performing its intended functions in its required manner.

P. Operating Cycle — Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

Q. 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.

R. 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.

Y. 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.

Z. Shutdown – The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations 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.

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

BB. Total Peaking Factor - The Total Peaking Factor (TPF) is the highest product of radial, axial, and local peaking factors simultaneously operative at any segment of fuel rod.

CC. 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.

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. When the reactor pressure is greater than 600 psig the combination of recirculation flow and reactor thermal power-to-water shall not exceed the limit shown in Figure 1.1.1. The safety limit is exceeded when the recirculation flow and thermal power-to-water conditions result in a point above or to the left of the limit line.

B. When the reactor pressure is less than 600 psig or recirculation flow is less than 5% of design, the reactor thermal power-to-water shall not exceed 460 MW(t).

C. 1. The neutron flux shall not exceed the scram setting established in Specification 2.1.A for longer than 1.5 seconds as indicated by the process computer.

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds

2.1 FUEL CLADDING INTEGRITY

Applicability:

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

Objective:

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

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

* 1. APRM - When the reactor mode switch is in the run position, the APRM flux scram setting shall be as shown in Figure 2.1.1 unless the combination of power and peak LHGR is above the curve in Figure 2.1.2. When the combination of power and peak LHGR is above the curve in Figure 2.1.2 a scram setting(s) as given by:

$$S \leq .65W + [55]\left[\frac{LTPF}{TPF}\right]$$

where:

S = setting in per cent of rated power
W = recirculation loop flow in per cent of rated flow

(Change #3*)

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING

TPF = LTPF unless the combination of power and peak LHGR is above the curve in Figure 2.1-2 at which point the actual peaking factor value shall be used.

LTPF = 3.05 for 7 x 7 fuel
LTPF = 3.01 for 8 x 8 fuel

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

1.1 SAFETY LIMIT

the scram setting established by Specification 2.1.A and a control rod scram does not occur.

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

2.1 LIMITING SAFETY SYSTEM SETTING

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

B. APRM Rod Block — The APRM rod block setting shall be as shown in Figure 2.1.1 unless the combination of power and peak LHGR is above the curve in Figure 2.1.2. When the combination of power and peak LHGR is above the curve in Figure 2.1.2 a rod block trip setting (S_{RB}) as given by:

$$S_{RB} \leq .65W + [43] \left[\frac{LTPF}{TPF}\right]$$

where:

the definitions used for the APRM scram trip apply.

C. Reactor Low Water Level Scram setting shall be ≥143" above the top of the active fuel at normal operating conditions.

D. Reactor Low Low Water Level ECCS initiation shall be 83" $(^{+4"}_{-0"})$ above the top of the active fuel at normal operating conditions.

E. Turbine Stop Valve Scram shall be ≤10% valve closure from full open.

F. Generator Load Rejection Scram shall initiate upon actuation of the fast closure solenoid valves which trip the turbine control valves.

G. Main Steamline Isolation Valve Closure Scram shall be ≤10% valve closure from full open.

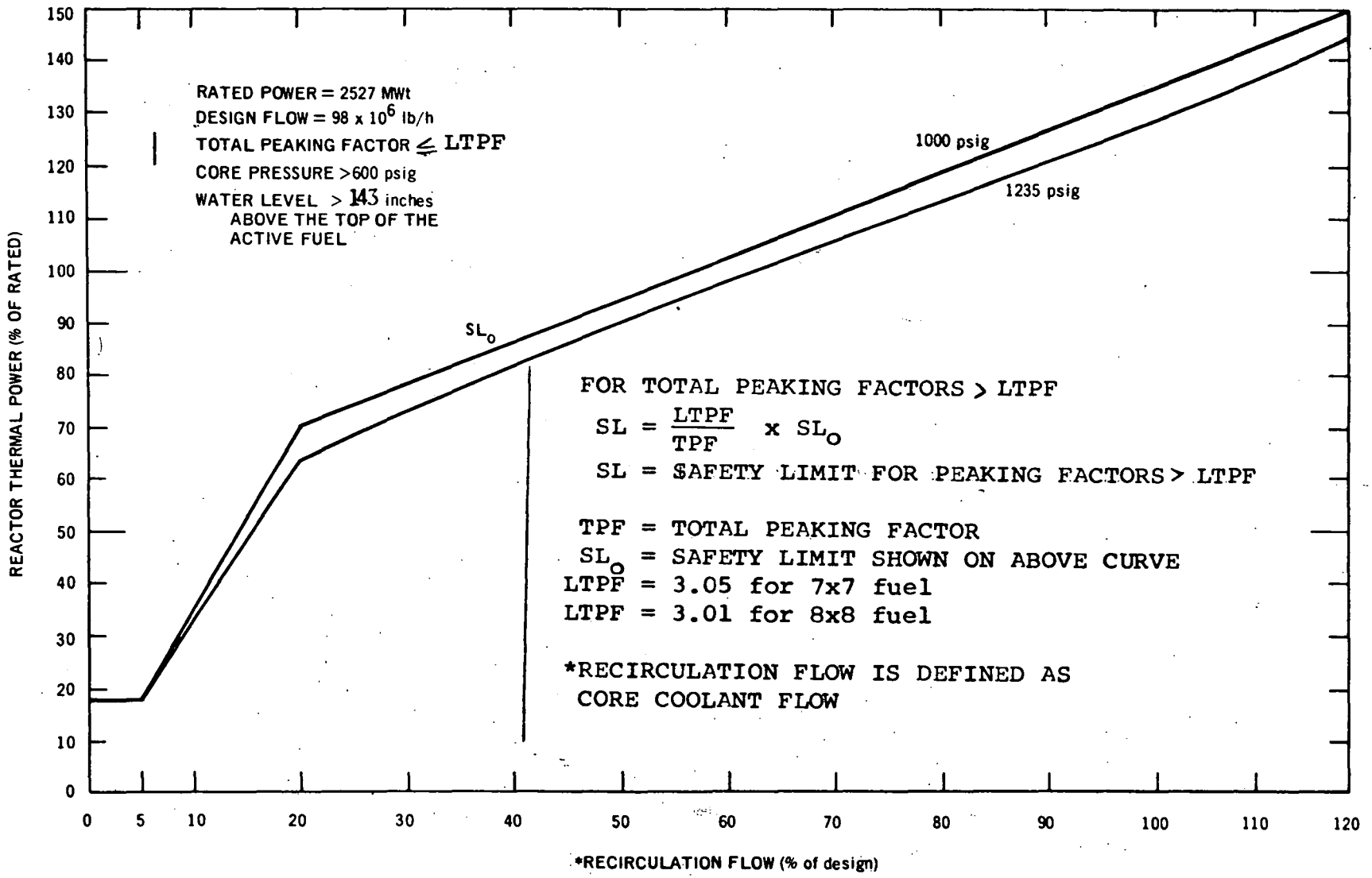

Figure 1.1.1. Core Thermal Safety Limit

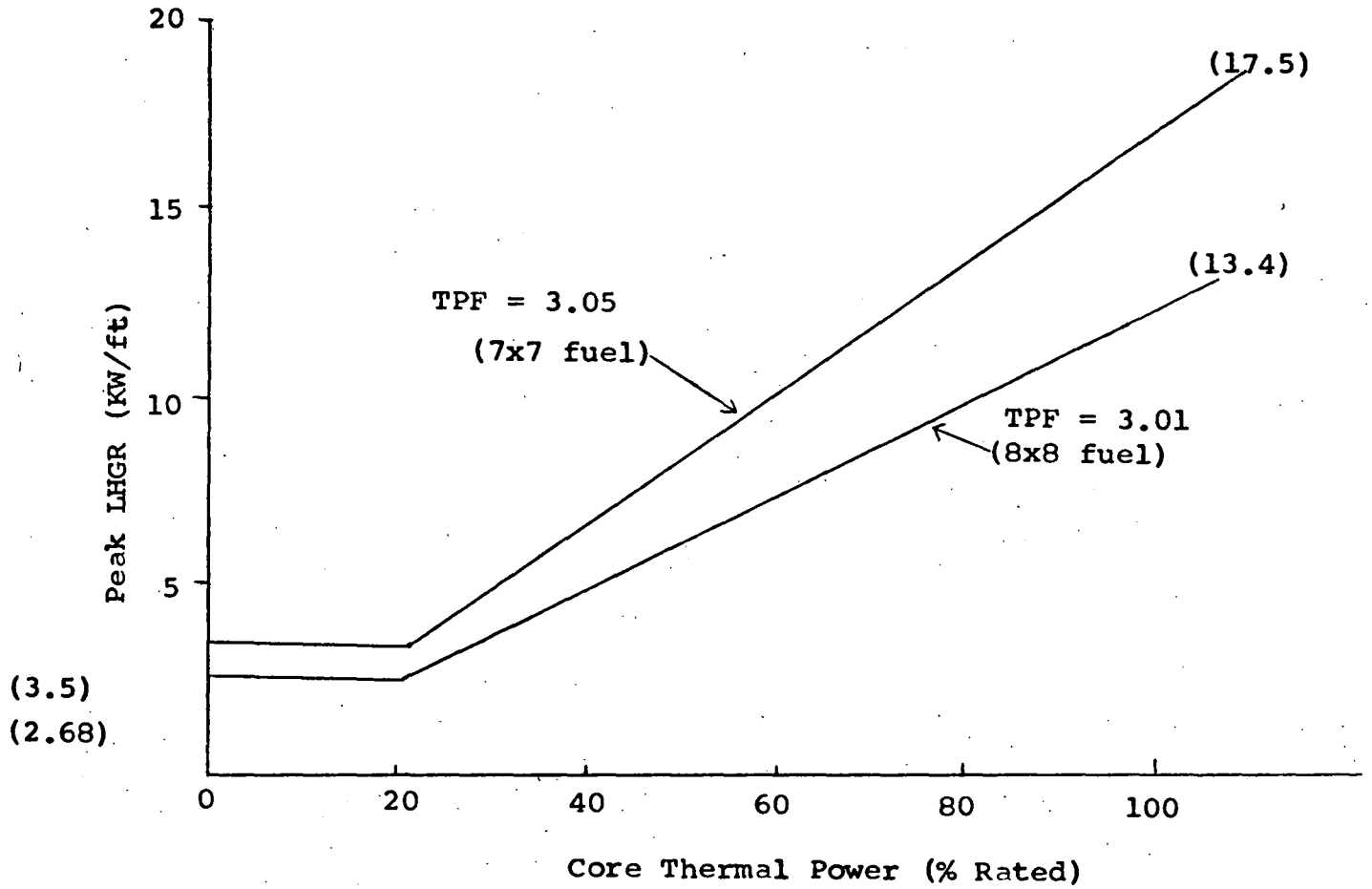

Figure 2.1-2 Peak LHGR Versus Core Thermal Power for a Limiting Total Peaking Factor

is based on a pressure of 1235 psig. In no case is reactor pressure ever expected to exceed 1250 psig, and therefore, the curves will cover all operating conditions with mere interpolation. If reactor pressure should ever exceed 1250 psig during power operation, it would be assumed that the safety limit has been violated. For pressures between 600 psig, which is the lowest pressure used in the critical heat flux data, and 1000 psig, the upper curve is applicable with increased margin.

The power shape assumed in the calculation of these curves was based on design limits and results **in a Limiting Total Peaking Factor. For any peaking of** smaller magnitude, the curves are conservative. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core Local Power Range Monitor (LPRM) System. However, to maintain applicability of the safety limit curve, the safety limit will be lowered according to the equation given on Figure 1.1.1 in the rare event of power operation with a total peaking factor in excess of **the Limiting Total Peaking Factor.**

The feedwater temperature assumed was the maximum design temperature output of the feedwater heaters at the given pressures and flows which is 348°F for rated thermal power. For any lower feedwater temperature, subcooling is increased and the curves are conservative.

The water level assumed in the calculation of the safety limit was that level corresponding to the bottom of the steam separator skirt (0" on the level instrument and approximately 12' above the top of the active fuel). This point is below the water level scram setpoint. As long as the water level is above this point the safety limit curves are applicable; i.e., the amount of steam carry under would not be increased and therefore the core inlet enthalpy and subcooling would not be influenced.

The values of the parameters involved in Figure 1.1.1 can be determined from information available in the control room. Reactor pressure and flow are recorded and the Average Power Range Monitor (APRM) in-core nuclear instrumentation is calibrated to read in terms of percent rated power.

The range in pressure and flow used for Specification 1.1.A was 600 psig to 1250 psig and 5% to 100%, respectively. Specification 1.1.B provides a requirement on power level when operating below 600 psig or 5% flow. In general, Specification 1.1.B will only be applicable during startup, hot standby, or shutdown of the plant. A review of all the applicable low pressure and low flow data (1, 2) has shown the lowest data point for transition boiling to have a heat flux of 144,000 Btu/hr/ft^2. To assure applicability to the Dresden 3 fuel geometry and provide some margin, a factor of 1/2 was used to obtain the critical heat flux; i.e., critical heat flux was assumed to occur for these conditions at 72,000 Btu/hr/ft^2. Assuming a peaking factor of 3.0, this is equivalent to a core average power of 460 MW(t) (18% of rated). This value is applicable to ambient pressure and no flow conditions. For any greater pressure or flow conditions, there is increased margin.

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8-9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed

---

(1) E. Janssen, "Multi-Rod Burnout at Low Pressure," ASME Paper 62-HT-26, August 1962.

(2) K. M. Becker, "Burnout Conditions for Flow of Boiling Water in Vertica Rod Clusters," AE-74 (Stockholm, Sweden), May 1962.

actually conducted from rated power but with the conservative void coefficient.

Inherent in these analyses is the fact that steady-state operation without forced recirculation flow will not be permitted except during startup testing.

In summary, the transients presented in the SAR were analyzed only up to the design flow control line and not above because:

1. The licensed maximum steady-state power level is 2527 MWt.

2. The units cannot physically be brought above 2527 MWt unless abnormal operation is employed.

3. Analyses of transients employ adequately conservative values of the controlling reactor parameters.

4. The analysis model itself is demonstrated to be conservative.

5. The analytical procedures now used result in a more logical answer than the alternative method of assuming a higher strating power, which has been shown above to be unrealistic, than using values for the parameters.

A. Neutron Flux Scram — The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent power. Since 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 transients with an APRM scram setting as shown in Figure 2.1.1, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analysis reported in the SAR demonstrates that, even with a fixed 120% scram trip setting, none of the postulated transients result in violation of the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-biased scram provides even additional margin. See page 15 for further comparison.

An increase in the APRM scram setting to greater than that shown in Figure 2.1.1 would decrease the margin present before the thermal hydraulic safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. A reduction in this operating margin would increase the frequency of spurious scrams which have an adverse affect on reactor safety because of unnecessary thermal stress which it causes. Thus, the APRM setting was selected because it provides adequate margin from the thermal hydraulic safety limit yet allows operating margin which minimizes unnecessary scrams.

The thermal hydraulic safety limit of Specification 1.1 was based on the Limiting Total Peaking Factor. A factor has been included on Figure 1.1.1 to adjust the safety limit in the event peaking factor exceeds the Limiting Total Peaking Factor. Likewise, the scram setting should also be adjusted to assure MCHFR does not become less than 1.0 in this degraded situation. This has been accomplished by use of Figure 2.1.2. If the combination of power and LHGR is greater than that shown by the curve, the APRM scram setting is adjusted downward by formula given in the specification. The scram setting as given by

the equation will prevent MCHFR from becoming less than 1.0 for the given heat flux condition for the worst expected transients. If the APRM scram setting should require a change due to an abnormal peaking condition, it will be done by changing the intercept point and thus, the entire flow bias curve will be shifted down. **Below 20% power the peak LHGR normally will be less than or equal to 20% power value. However, if the peak LHGR below 20% power exceeds the 20% power value, the APRM scram and rod block settings shall be lowered by the formula in the specifications. The above safety margins are not significantly reduced because power maneuvers below 20% power are restricted to control rod movements due to the protective interlocks limiting recirculation pump operation to minimum speed. During this period flow increases inherently occur with power increases, even with no recirculation pumps operating. Pump operation enhances this phenomenon at minimum pump speed. Since TPF improves with nearly every rod withdrawal, and power ascension must be accomplished by slow rod withdrawal, the specification provides operational flexibility while still maintaining adequate margin to the Safety Limit.**

* For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 18% 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. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and
* the APRM system would be more than adequate to assure a scram before the power could exceed
* the safety limit. The 15% 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.

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

* The IRM system consists of 8 chambers, 4 in each of the reactor protection system logic channels, arranged in the core as shown in Figure 7.4.4 of the FSAR. 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 covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges,

each being 1/2 of a decade in size. The IRM scram 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 at 120 divisions for that range; likewise, if the instrument were on range 5, the scram setting would be 120 divisions on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram setting is also ranged up. In the start-up/hot standby mode, a scram at 120 divisions on the instrument is less than 15% power, except for range 10 on the instrument. Thus, the scram setting on the IRM is also less than the 15% scram on the APRM, except in the 10th range. The IRM scram provides
* protection for changes which occur, both locally and over the entire core. The IRM, because of the scram arrangement discussed above, thus provides additional or back-up protection to the APRM 15% scram in the start-up and hot standby mode. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit or the APRM 15% scram occurred. For the case of a single control rod withdrawal error this transient has been analyzed in Section 7.4.4.3 of the FSAR. 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. This condition exists at quarter rod density. Quarter rod density is illustrated in Section 7.4.5 of the FSAR. 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 1% of rated power, thus maintaining heat flux within those values specified in the safety limit for this condition of plant operation. 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 back-up protection for the APRM.

B. APRM Control Rod Block Trips — Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against exceeding a MCHFR of unity. This rod block setpoint, which is automatically varied with recirculation flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The specified flow variable setpoint provides substantial margin from fuel damage, assuming steady-state operation at the setpoint, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip point vs. flow relationship; therefore, the worst case MCHFR during steady-state operation is at 108% of rated power. Peaking factors as specified in Section 3.2.5 of the SAR were considered. The total peaking factor was 3.0. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram setting, the APRM rod block setting is adjusted downward if peaking factors are greater than the Limitir- Total Peaking Factor. This assures rod block will occur before MCHFR becomes less than 1.0 even for this degraded case. The rod block setting is changed by changing the intercept point of the flow bias curve; thus, the entire curve will be shifted downward.

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 increase 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 will not be set lower because of ECCS capacity requirements.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS 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.

3.1 LIMITING CONDITION FOR OPERATION	4.1 SURVEILLANCE REQUIREMENT

3.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

To assure the operability of the reactor protection system.

Specification:

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The response times of the individual functions shall not exceed 0.10 second.

B. During operation with a Limiting Total Peaking Factor, either:

   a. The APRM scram and rod block settings shall be reduced to the values given by the equations in Specifications 2.1.A.1 and 2.1.B; or

   b. The power distribution shall be changed such that a Limiting Total Peaking Factor no longer exists.

4.1 REACTOR PROTECTION SYSTEM

Applicability:

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

Objective:

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

Specification:

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

B. Daily during reactor power operation, the peak LHGR shall be determined and the core power distribution shall be checked for Limiting Total Peaking Factor.

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

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

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

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

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

Experience with passive type instruments in Commonwealth Edison generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less than 0.4%/month; i.e., in the period of a month a drift of .4% would occur and thus providing for adequate margin.

For the APRM system drift of electronic apparatus is not the only consideration in determining a calibration frequency. Change in power distribution and loss of chamber sensitivity dictate a calibration every seven days. Calibration on this frequency assures plant operation at or below thermal limits.

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

B. The peak LHGR shall be checked once per day to determine if the APRM 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 peak LHGR is adequate.