DOCKET FILE

50-237

January 21, 1971

Commonwealth Edison Company
ATTN: Mr. Byron Lee, Jr.
Assistant to the President
P.O. Box 767
Chicago, Illinois 60690

Change No. 8
License No. DPR-19

Gentlemen:

We have reviewed your Proposed Change No. 8, dated January 19, 1971, requesting changes to Sections 1.1 and 2.1 and to Table 3.2.3 of the Technical Specifications of Provisional Operating License No. DPR-19. The proposed changes would extend the core thermal safety limit to 120% of design recirculation flow and change the slope of the APRM flow biased scram and rod block curves. These changes will make the corresponding sections of the Technical Specifications for Dresden Nuclear Power Station Units 2 and 3 identical.

We have concluded that the proposed change does not present significant hazards considerations not described or implicit in the safety analysis report and that there is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, pages 6-23 of Sections 1.1 and 2.1 are replaced by the enclosed pages 6 (TS-8) - 19 (TS-8), and Table 3.2.3 is replaced by the enclosed Table 3.2.3 (TS-8).

As discussed with you during the recent review of Dresden Unit 3, there are other sections of the Dresden Unit 2 Technical Specifications that should be modified to assure consistency between Dresden Units 2 and 3 Technical Specifications. We will be happy to discuss these areas with you.

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

Enclosures:
1. Pages 6 (TS-8) - 19 (TS-8) of Sections 1.1 and 2.1
2. Table 3.2.3 (TS-8)

cc: Athur Gehr, Esquire
Isham, Lincoln & Beale

22121

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T31 R1	SURNAME ▶	GCLainas:mlm	RTedesco	RSBoyd	RSchroeder	PAMorris
	DATE ▶	1/21/71	1/21/71	1/21/71	1/21/71	1/21/71

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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cc: Athur Gehr, Esquire
Isham, Lincoln & Beale

Dup 2-2-7

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 – The APRM flux scram setting shall be as shown in Figure 2.1.1 unless the combination of power and peak heat flux is above the curve in Figure 2.1.2. When the combination of power and peak heat flux is above the curve in Figure 2.1.2 a scram setting(s) as given by:

$$S = \frac{502,000\ P}{X}$$

where:

P = percent of rated power
X = peak heat flux – ($Btu/hr/ft^2$)

2. IRM – The IRM flux scram setting shall be $\leq 15\%$ of rated neutron flux.

1.1 SAFETY LIMIT	2.1 LIMITING SAFETY SYSTEM SETTING

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.

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 heat flux is above the curve in Figure 2.1.2. When the combination of power and peak heat flux is above the curve in Figure 2.1.2 a rod block trip setting (S_{RB}) as given by:

$$S_{RB} = \frac{451,440\ P}{X}$$

where:

P = percent of rated power
X = peak heat flux (Btu/hr/ft^2)

shall be used.

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.

H. Main Steamline Pressure initiation of main steamline isolation valve closure shall be ≥850 psig.

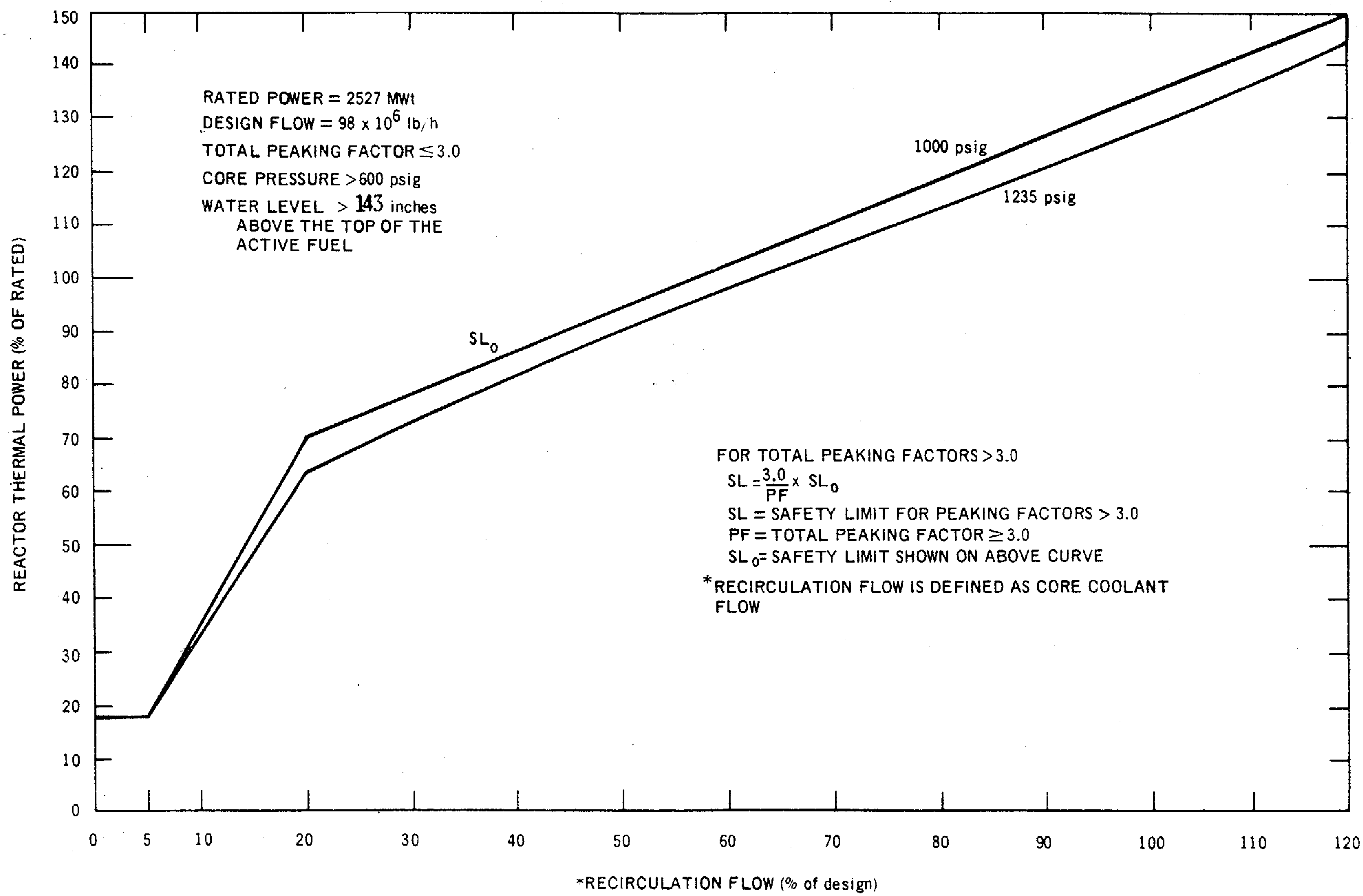

Figure 1.1.1. Core Thermal Safety Limit

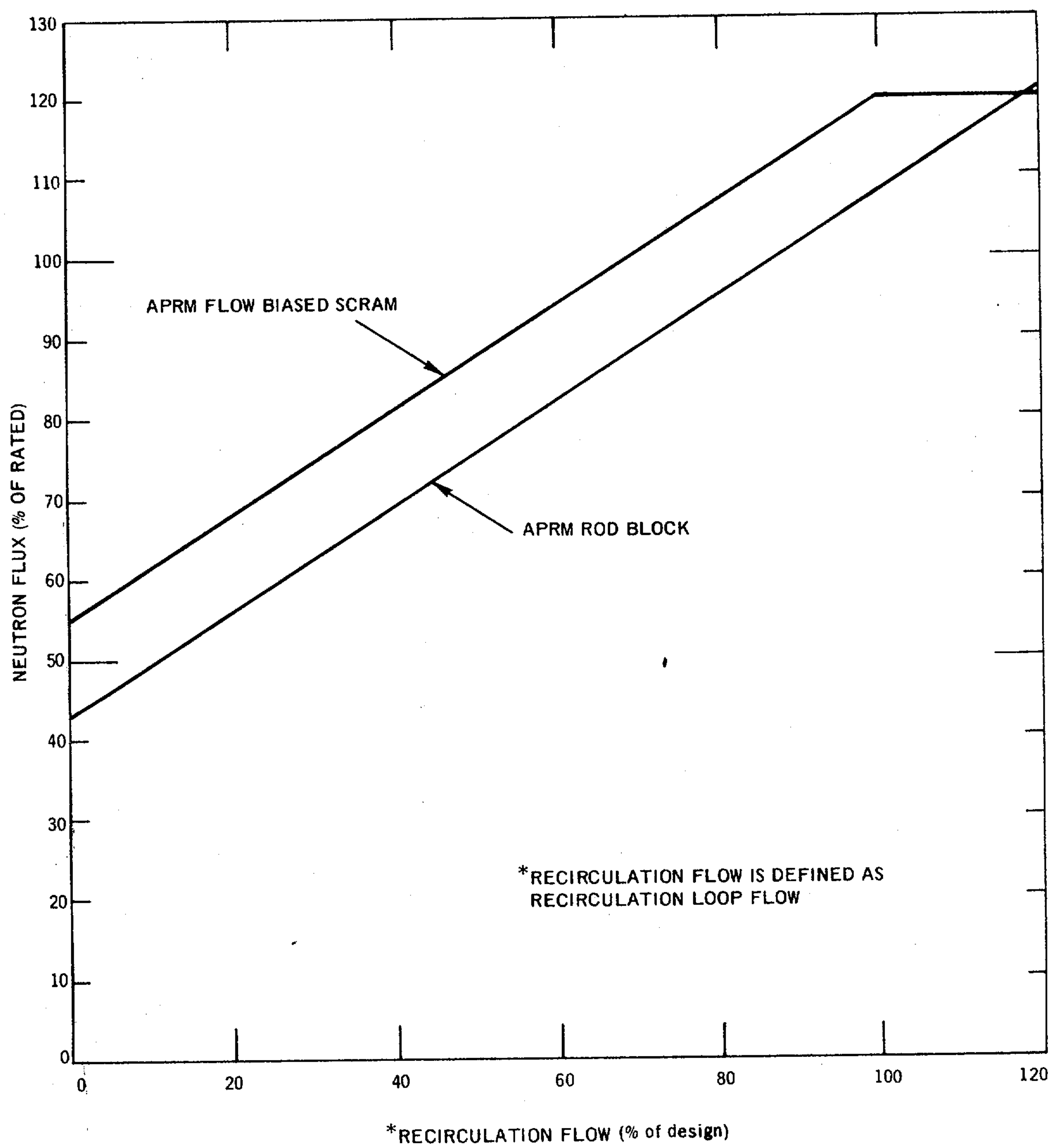

Figure 2.1.1. APRM Flow Reference Scram and APRM Rod Block Settings

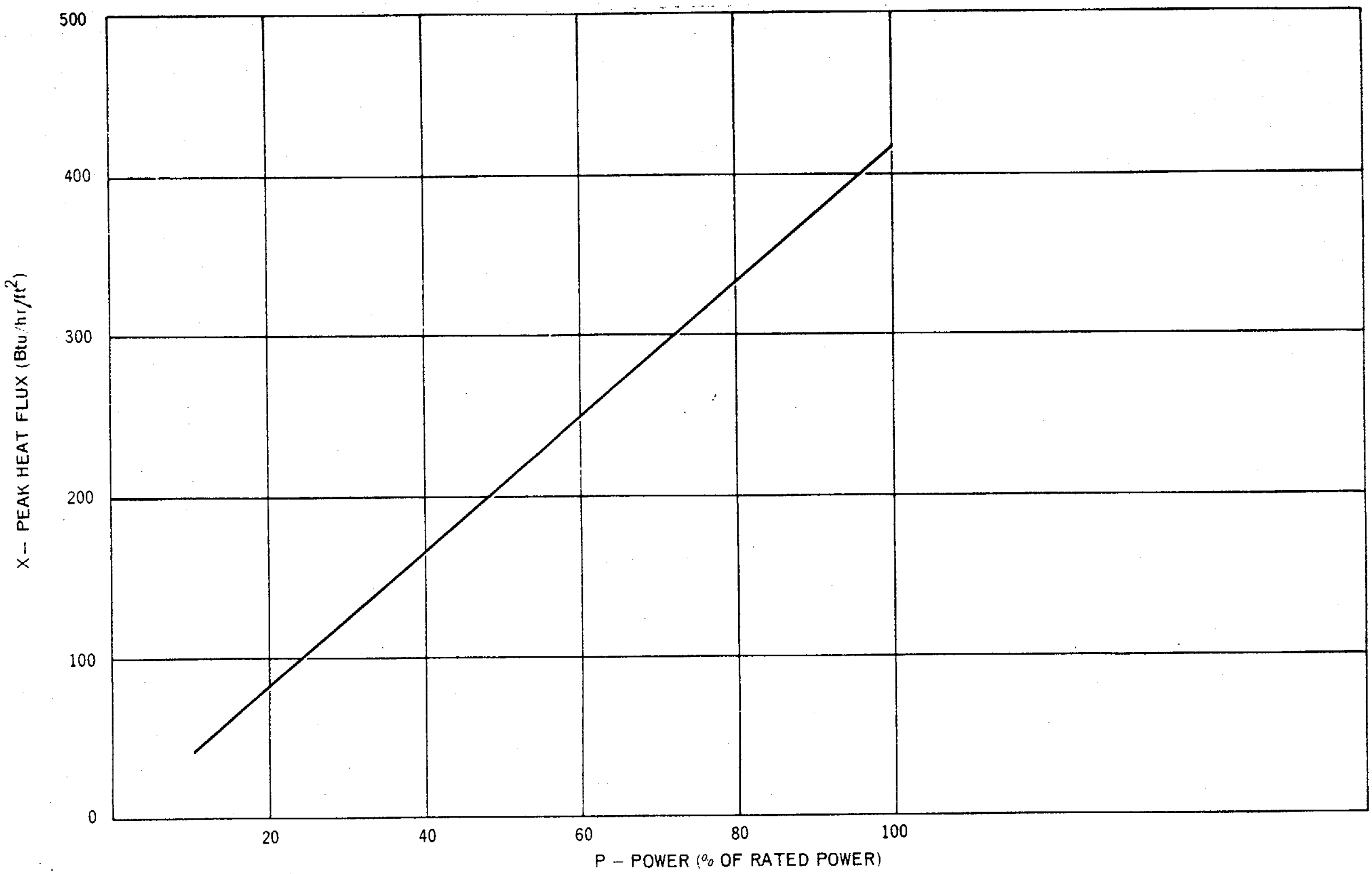

Figure 2.1.2. APRM Scram and Rod Block Setting for Total Peaking Factors > 3.0

Bases:

1.1 The fuel cladding represents one of the physical barriers which separate radioactive materials from environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system setpoints. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally-caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined in terms of the reactor operating conditions which can result in cladding perforation.

Departure from nucleate boiling (DNB) results in a decrease in heat transfer from the clad and, therefore, elevated clad temperatures and the possibility of clad failure. However, the existence of a critical heat flux (CHF) or departure from nucleate boiling, is not a directly observable parameter in an operating reactor. Furthermore, the critical heat flux correlation data which relates observable parameters to the magnitude of critical heat flux is statistical in nature. The safety limit represented in Figure 1.1.1 is based on the significant observable parameters involved in the critical heat flux correlation and is taken at the core design critical heat flux correlation level of confidence that a critical heat flux occurrence will not occur on any fuel rod in the core during normal plant operation, including anticipated transients.

The safety limit curves shown in Figure 1.1.1 represent the locus of operation conditions for which the maximum powered rod has a minimum critical heat flux ratio (MCHFR) equal to 1.0. The MCHFR value was determined using the design basis critical heat flux correlation given in APED 5286[1]. The operating range with MCHFR >1.0 is below and to the right of these curves. The evaluation is based on the operating map as given in Figure 3.2.3 of the SAR.

The design basis critical heat flux is based on an interrelationship of reactor coolant flow and steam quality. Steam quality is determined by reactor power, pressure and coolant inlet enthalpy which in turn is a function of feedwater temperature and to a lesser degree reactor water level. This correlation is based upon experimental data taken over the pressure range of interest in a BWR, and the correlation line was very conservatively drawn below all the available data. Since the correlation line was drawn below the data, there is a very high probability that operation at the calculated safety limit would not result in a critical heat flux occurrence. In addition, if a critical heat flux were to occur, clad perforation would not necessarily be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Dresden 3 operated above the critical heat flux for significant period of time (30 minutes) without clad perforation.

Curves are presented for two different pressures in Figure 1.1.1. The upper curve is based on a nominal operating pressure of 1000 psig. The lower curve

(1) J. M. Healzer, J. E. Hench, E. Janssen, S. Levey, Design Basis for Critcal Heat Flux Condition in Boiling Water Reactors, APED 5286, General Electric, San Jose, California, September, 1966.

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 total peaking factor of 3.0. 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 3.0.

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

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(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.

in detail (3). In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked each refueling outage to assure the insertion times are adequate. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analysis show that even if the bypass system fails to operate, the design limit of MCHFR = 1.0 is not exceeded. Thus, use of a 1.5 second limit provides additional margin.

The computer provided with Dresden Units 2 and 3 has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C.2 will be relied on to determine if a safety limit has been violated.

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

The proposed fuel operating conditions for Unit 3 reflect linear power generation rates and exposures higher than those experienced previously in BWR plants. Additional experimental data beyond that presented in Amendment 15 of the SAR will be obtained to further support the proposed combinations of fuel linear power generation rates and exposures, considering both normal and anticipated transient modes of operation. To develop these data for further assurance of fuel integrity under all modes of plant operation, a surveillance program on BWR fuel which operates beyond current production fuel experience will be undertaken. The schedule of inspections will be contingent on the availability of the fuel as influenced by plant operating and facility requirements. The program, as outlined in Amendment 17 of the SAR, will include surveillance of reactor plant off-gas activity, relevant plant operating data and fuel inspection.

(3) SAR, Section 4.4.3 for turbine trip and load reject transients, Section 4.3.3 for flow control full coupling demand transient, and Section 11.3.3 for maximum feedwater flow transient.

Bases:

2.1 The transients expected during operation of the Dresden 3 unit have been analyzed starting at the rated thermal power condition of 2527 MWt at 100% recirculation flow. It should be noted that this power is equivalent to the designed maximum power and a higher power cannot physically be obtained under normal operating conditions unless the turbine bypass system is used. In addition, 2527 MWt is the licensed maximum steady-state power level of Dresden 3. This maximum steady-state power will never be knowingly exceeded.

Dresden 3 was not analyzed from a power level which included instrument errors. To protect against misleading conclusions from analysis not reflecting realistic instrument errors, conservatism was incorporated by conservatively estimating the controlling factors such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, axial power shapes, etc. These factors are all selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for the evaluation of reactor dynamics performance. Comparisons have been made showing results obtained from a General Electric boiling water reactor and the predictions made by the model. The comparisons and results are summarized in Topical Report APED-5698, "Summary of Results Obtained From A Typical Startup and Power Test Program for a General Electric Boiling Water Reactor."

The void reactivity coefficient utilized in the analysis is conservatively estimated to be about 25% larger than the most negative value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to the scram worth of about 75% of the control rods. The scram delay time and rate of rod insertion are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The insertion of the first dollar of reactivity strongly turns the transient and the stated 10% insertion time conservatively accomplishes this desired initial effect. The time for 50% and 90% insertion are given to assure proper completion of the insertion stroke, to further assure the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

The design peaking factors at the full power conditions for Dresden 3 result in a MCHFR value of 2.04. For analysis of the thermal consequences of the transients, higher peaking factors are used, such that a MCHFR of 1.9 is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the rated power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels. As an example, consider the sensitivity analyses conduct to provide the answer to Question 4.6.4 of Amendment 7 of the Dresden Unit 2 SAR. From the results of the Case 1 transient, the turbine trip with flux scram without bypass or relief, a significant reduction in the neutron flux and heat flux peaks will be realized when the smaller void reactivity coefficient is used. For this particular transient, if it were also analyzed at a power level of 110% of rated but with the expected void reactivity coefficient, the resulting heat flux peak would be less than the peak resulting from the analysis 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 a total peaking factor of 3.0. A factor has been included on Figure 1.1.1 to adjust the safety limit in the event peaking factor exceeds 3.0. 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 heat flux is greater than that shown by the curve; i.e., a peaking factor greater than 3.0 exists, the APRM scram setting is adjusted downward by formula given in the specification. The scram setting as given by

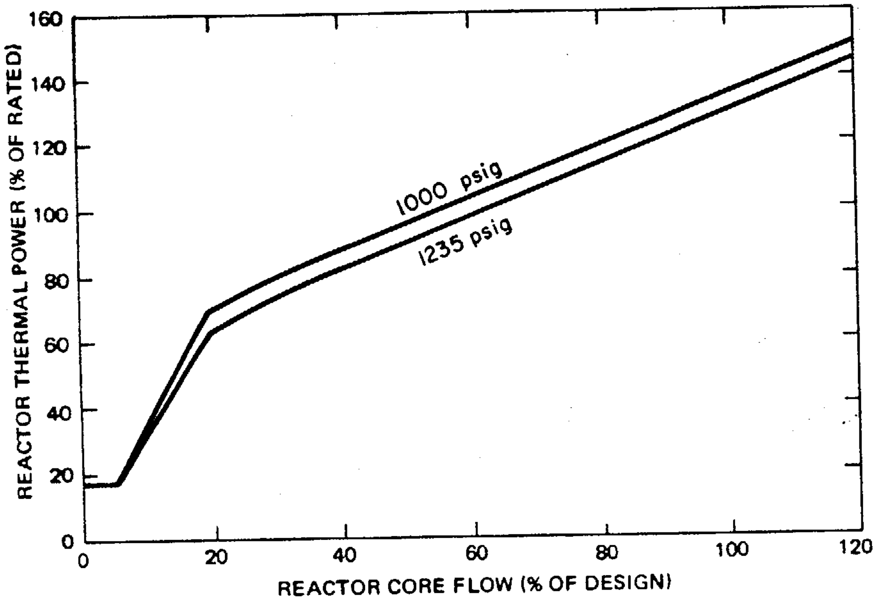

FIGURE A

Core Thermal Safety Limit vs Reactor Core Flow

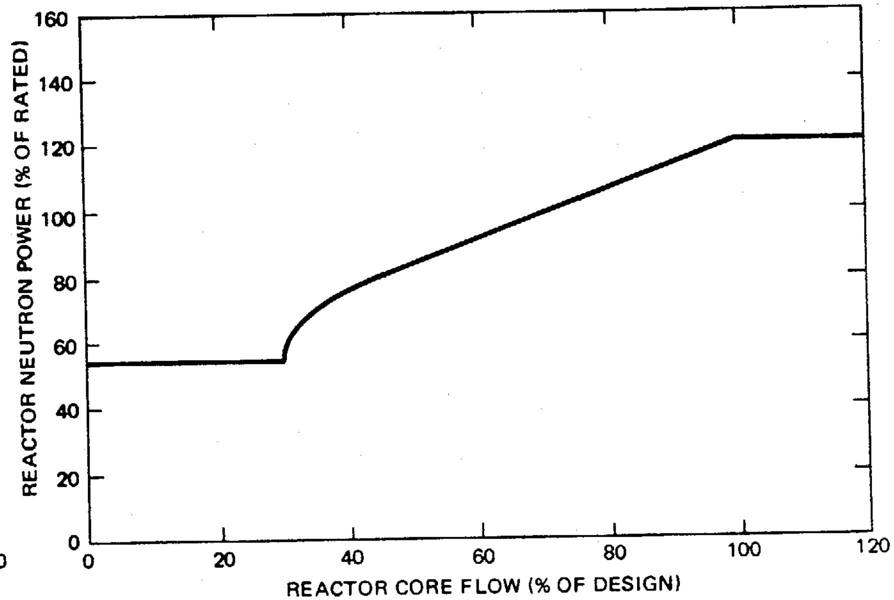

FIGURE B

APRM Flow Bias Scram vs Reactor Core Flow

Figure A above is a copy of the Core Thermal Safety Limit which is shown in more detail on Figure 1-1.1. Figure B above represents the APRM flow bias scram with neutron flux plotted against reactor core flow instead of reactor recirculation loop flow as shown in Figure 2.1.1. During steady state power operation of the reactor the neutron power will be equal to the thermal power and the two curves, Figures A and B can be compared.

It should be remembered, however, that during transient operation the heat flux (thermal power) lags behind the neutron flux due to the inherent heat transfer time constant of the fuel. During such transient operation the two curves, Figures A and B are not comparable.

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.

For operation in the startup mode while the reactor is at low pressure, the IRM 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 IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM 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.

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 greater than 3.0 exist. This assures a 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.

E. Turbine Stop Valve Scram — The turbine stop valve scram like the load rejection scram anticipates the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valves and failure of the bypass. With a scram setting at 10% of valve closure the resultant increase in surface heat flux is the same as for the load rejection and thus adequate margin exists. No perceptable change in MCHFR occurs during the transient. Ref. Section 11.2.3 SAR.

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 MCHFR from becoming less than 1.0 for this transient. For the load rejection from 100% power, the heat flux increases to only 106.5% of its rated power value which results in only a small decrease in MCHFR. Ref. Section 4.4.3 SAR.

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

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

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TABLE 3.2.3 (TS-8)

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)	$\leq 0.650W + 43$ (2)
2	APRM downscale (7)	$\geq 3/125$ full scale
1	Rod block monitor upscale (flow bias) (7)	$\leq 0.650W + 45$ (2)
1	Rod block monitor downscale (7)	$\geq 5/125$ full scale
3	IRM downscale (3)	$\geq 5/125$ full scale
3	IRM upscale	$\leq 108/125$ full scale
2 (5)	SRM detector not in startup position	(4)
2 (5) (6)	SRM upscale	$\leq 10^5$ counts/sec

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 and IRM downscale need not be operable in the "Run" position and APRM downscale, APRM upscale, RBM upscale, and RBM downscale need not be operable in the Startup/Hot Standby mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.
2. W is the reactor recirculation loop flow in percent. Trip level setting is in percent of full power.
3. IRM downscale may be bypassed when it is on its lowest range.
4. This function may be bypassed when the count rate is ≥ 100 cps.
5. One of the four SRM inputs may be bypassed.
6. This SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.
7. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).