2.0 SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

2.1 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 reactor coolant core flow and reactor thermal power transferred to the coolant shall not exceed the limit shown in Figure 2.1.1. The safety limit is exceeded when the reactor coolant core flow and thermal power transferred to the coolant results in a point above or to the left of the limit line.

2.3 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 trip setting shall be as shown in Figure 2.3.1 unless the combination of power and peak heat flux is above the applicable curve in Figure 2.3.2. When the combination of power and peak heat flux is above the curve in Figure 2.3.2, a scram setting (S) as given by:

B. When the reactor pressure is less than 600 psig or core flow is less than 5% of design, the reactor thermal power transferred to the coolant shall not exceed 300 MW.

C. 1. The neutron flux shall not exceed the scram setting established in Specification 2.3.A for longer than 0.95 seconds as indicated by the process computer.

$$S = \frac{485{,}000\ P}{X} \quad \text{(7x7 fuel)}$$

$$S = \frac{425{,}000\ P}{X} \quad \text{(8x8 fuel)}$$

Where:
P = percent of rated power
X = peak heat flux - ($BTU/HR/FT^2$)
shall be used.

2. IRM--Flux Scram setting shall be ≤ 15% of rated neutron flux.

B. APRM Rod Block - The APRM rod block setting shall be as shown in Figure 2.3.1 unless the combination of power and peak heat flux is above the applicable curve in Figure 2.3.2. When the combination of power and peak flux is above the curve in Figure 2.3.2, a rod block trip setting (RB) as given by:

$$RB = \frac{437{,}400\ P}{X} \quad \text{(7x7 fuel)}$$

$$RB = \frac{382{,}400\ P}{X} \quad \text{(8x8 fuel)}$$

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 ≥ 10'6 above the top of the active fuel.

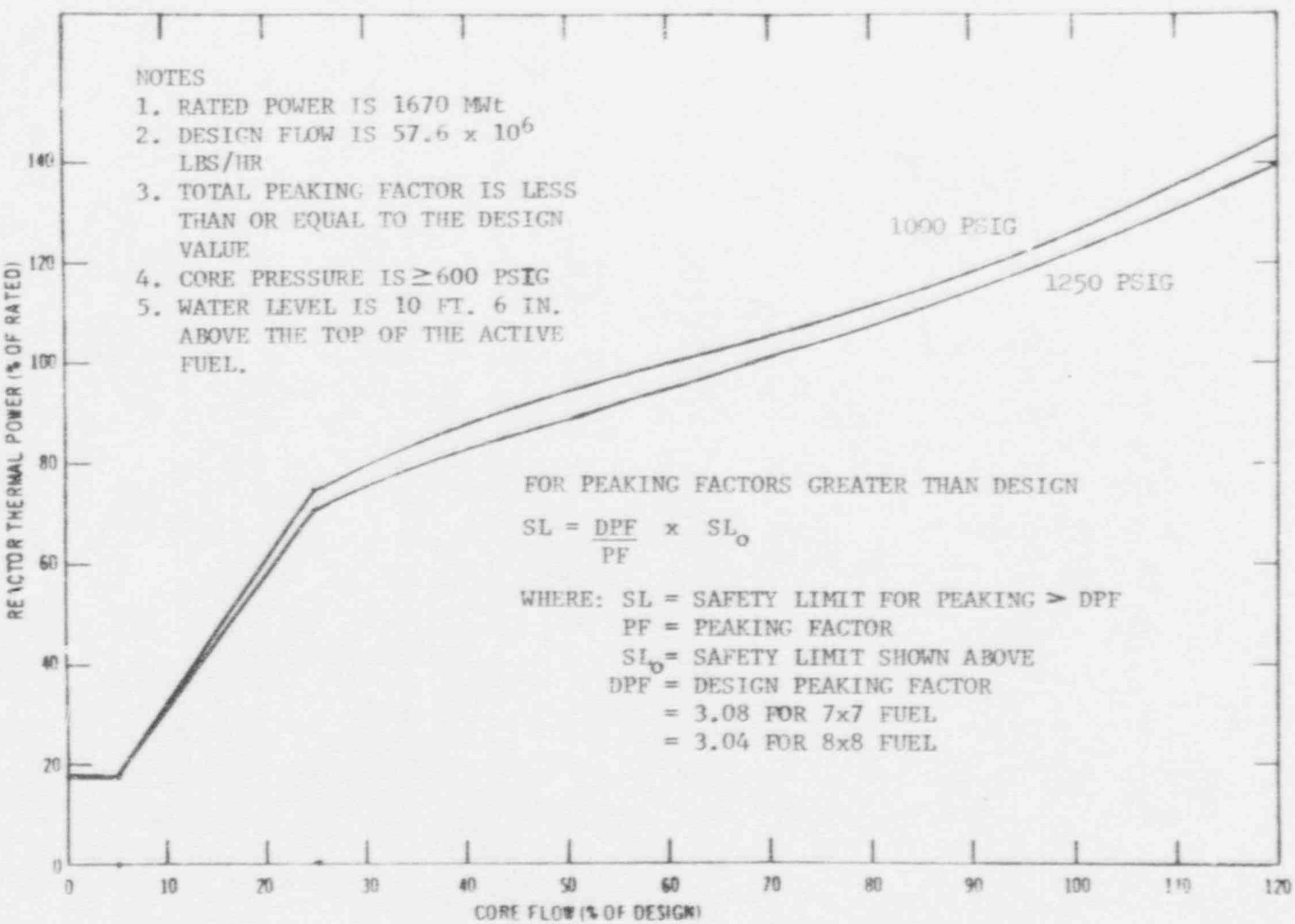
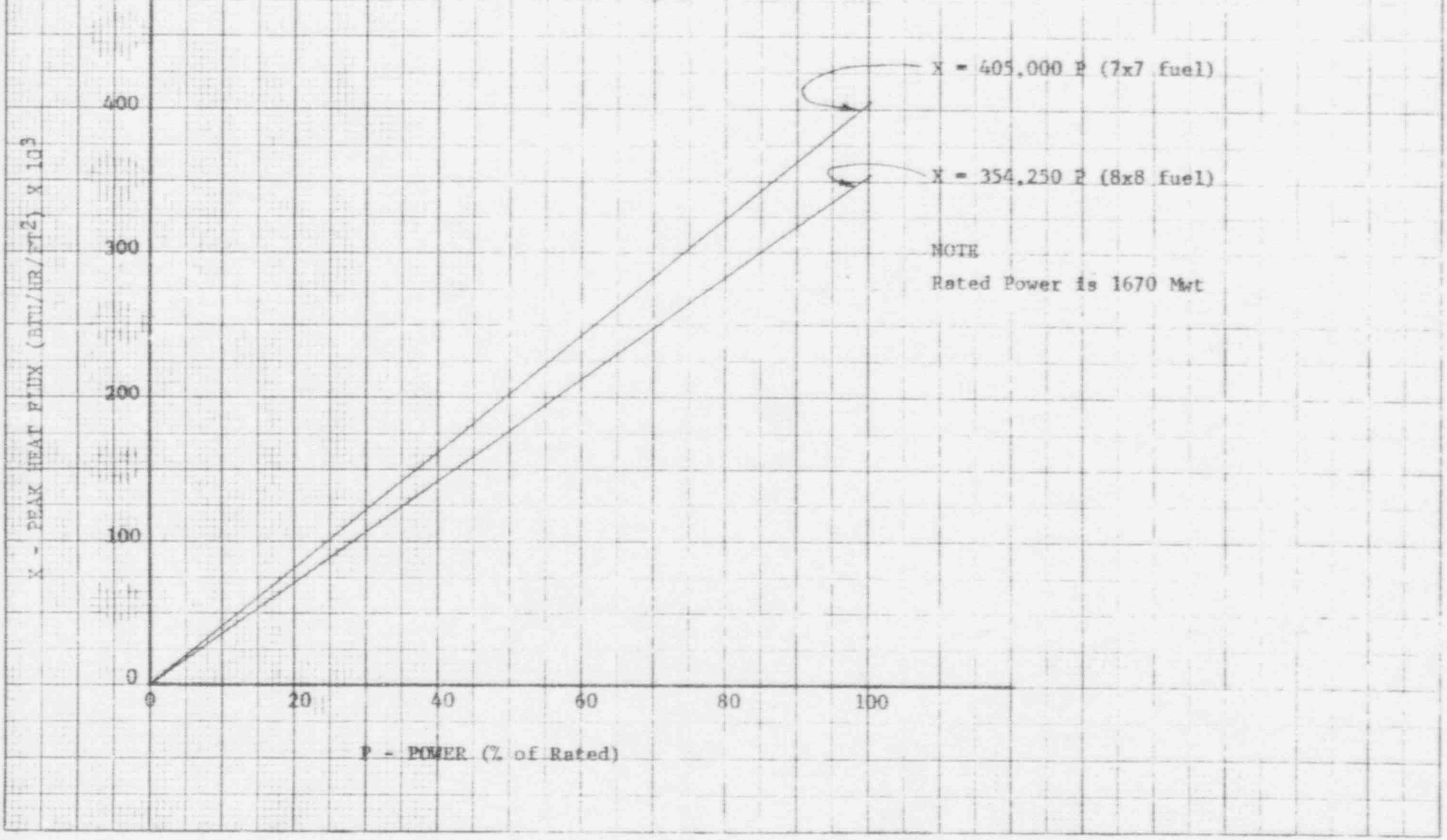


FIGURE 2.1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT

FIGURE 2.3;2

RELATIONSHIP BETWEEN PEAK HEAT FLUX AND POWER FOR PEAKING FACTORS OF 3.08 (7x7 fuel) AND 3.04 (8x8 fuel)



Bases Continued:

2.1 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 [illegible] operated above the critical heat flux for significant period of time (30 minutes) without clad perforation. (1)

Curves are presented for two different pressures in Figure 2.1.1. The upper curve is based on a normal operating pressure of 1000 psig. The lower curve is based on a pressure of 1250 psig. In no case is reactor pressure ever expected to exceed 1250 psig, and therefore, the curves will cover all operating conditions with 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 3.08 for 7x7 fuel and 3.04 for 8x8 fuel. 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. To maintain applicability of the safety limit curve, the safety limit will be lowered according to the equation given in Figure 2.1.1 in the rare event of power operation with a total peaking factor in excess of the design value.

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(1) T. Sorlie, et. al. - "Experiences with Operating BWR Fuel Rods above the Critical Heat Flux" - Nucleonics, Vol. 23, No. 4, April, 1965.

2.1 During transient operation, the heat flux would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel. 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 in detail (4,5,6,7). 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 delay in the control rod action to reduce neutron flux to less than the scram setting within 0.95 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed anytime a neutron flux scram setting of the APRM's is exceeded for longer than 0.95 seconds.

Analysis within the nominal uncertainty range of all appropriate significant parameters, show that if the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 0.95 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected.

The computer provided with Monticello 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 set point 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 2.1.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.

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(4) FSAR Volume I, Section III-2.2.3
(5) FSAR Volume III, Sections XIV-5
(6) Supplement on Transient Analyses submitted by NSP to the AEC February 13, 1973
(7) Letter from NSP to the AEC, "Planned Reactor Operation from 2,000 MWD/T to end of cycle 2", dated August 21, 1973

Bases Continued:

2.3 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 per cent 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 induced by disturbances and with an APRM scram setting as shown in Figure 2.3.1, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analysis reported in the FSAR 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 additional margin.

An increase in the APRM scram setting to greater than that shown in Figure 2.3.1, 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 former 120% APRM setting was selected because it provides adequate margin from the thermal-hydraulic safety limit, yet allows operating margin which minimizes the possibility of unnecessary scrams. Therefore, it is intended to ultimately replace (with prior AEC approval) the automatic flow referenced scram with a fixed 120 percent scram setting, providing that initial power operation confirms the nuclear behavior characteristics used in these transient analysis.

The thermal hydraulic safety limit of Specification 2.1 was based on a total peaking factor of 3.08 for 7x7 fuel and 3.04 for 8x8 fuel. A factor has been included on Figure 2.1.1 to adjust the safety limit in the event peaking factor exceeds the design value. 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 2.3.2. If the combination of power and heat flux is greater than shown by the curve; i.e., a peaking factor greater than the design value exists, the APRM scram setting is adjusted downward by the 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 increasing the APRM gain and thus reducing the slope and intercept point of the flow - biased scram curve by the reciprocal of the APRM gain change.

Bases Continued:

2.3 the worst case MCHFR during steady state operation is at 110% of rated power. Peaking factors as specified in Section 3.2 of the FSAR were considered. The total peaking factor was 3.08 for 7x7 fuel and 3.04 for 8x8 fuel. 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 the design value 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 increasing the APRM gain and thus reducing the slope and intercept point of the flow-biased rod block curve by the reciprocal of the APRM gain change.

The operator will set the APRM rod block trip settings no greater than that shown in Figure 2.3.1. However, the actual set point can be as much as 3% greater than that shown on Figure 2.3.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on Page 18.

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 operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual set point can be as much as 6 inches lower due to the deviations discussed on Page 18.

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%. The design of the ECCS components to meet the above criterion was dependent on three previously set parameters: the maximum break size, the low water level scram set point, and the ECCS initiation set point. To lower the set point for initiation of the ECCS could prevent the ECCS components from meeting their criterion. To raise the ECCS initiation set point 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.

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