1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

1. <u>APRM Flux Scram Trip Setting (Run Mode)</u>

When the reactor mode switch is in the run position, the APRM flux scram setting shall be:

$$S \leq \left[.65W_D + 55\right]$$

with a maximum set point of 120% for core flow equal to 98 x 10^6 lb/hr and greater.

Where:

S = setting in per cent of rated power

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

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (.65W_D + 55) \left[\frac{FRP}{MFLPD} \right]$$

Where:

FRP = fraction of rated thermal power (2527 Mwt)

MFLPD = maximum fraction of limiting power density where the limiting power density for each bundle is the design linear heat generation rate for that bundle.

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. <u>APRM Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)</u>

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

1.1 SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psig)

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

C. Power Transient

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

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.A and a control rod scram does not occur.

D. Reactor Water Level (Shutdown Condition)

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

*Top of active fuel is defined to be 360 inches above vessel zero (see [illegible]

2.1 LIMITING SAFETY SYSTEM SETTING

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:

$$S \leq \left[.65W_D + 43\right]$$

The definitions used above for the APRM scram trip apply.

In the event of operation with a maximum fraction limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (.65W_D + 43)\left\{\frac{FRP}{MFLPD}\right\}$$

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. 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.

B. Core Thermal Power Limit (Reactor Pressure < 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 28×10^3 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 with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. At 25% of rated thermal power, the peak powered bundle would have to be operating at 3.84 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

C. Power Transient

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 in detail. 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.

Control rod scram times are checked as required by specification 4.3.C. 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 analyses show that even if the bypass system fails to operate, the design limit of MCPR = the MCPR fuel cladding integrity safety limitation exceeded. Thus, use of a 1.5 second limit provides additional margin.

INSTRUMENTATION THAT INITIATES ROD BLOCK

Table 3.2.3

Minimum No. of Operable Inst. Channels Per Trip System(1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias)(7)	$\le \left[.65W_D + 43\right] \frac{FRP}{MFPLD}$ (2)
* 1	APRM upscale (refuel and Startup/Hot Standby mode)	$\le$ 12/125 full scale
2	APRM downscale (7)	$\ge$ 3/125 full scale
1	Rod block monitor upscale (flow bias)(7)	$\le \left[.65W + 42\right]$ (2)
1	Rod block monitor downscale (7)	$\ge$ 5/125 full scale
3	IRM downscale (3)	$\ge$ 5/125 full scale
3	IRM upscale	$\le$ 108/125 full scale
* 3	IRM detector not fully inserted in the core	
2(5)	SRM detector not in startup position	(4)
2(5)(6)	SRM upscale	$\le 10^5$ counts/sec

Bases:

3.2 In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminates operator errors before they result in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, control rod block and standby gas treatment systems. The objectives of the specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiates or controls core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. It should be noted that the setpoints of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low-reactor water level instrumentation is set to trip at >8 inches on the level instrument (top of active fuel is defined to be 360 inches above vessel zero) and after allowing for the full power pressure drop across the steam dryer the low level trip is at 504 inches above vessel zero, or 144 inches above the top of active fuel. Retrofit 8x8 fuel has an active fuel length 1.24 inches longer than earlier fuel designs. However, present trip setpoints were used in the LOCA analyses (NEDO-24146A, April 1979). This trip initiates closure of Group 2 and 3 primary containment isolation valves but does not trip the recirculation pumps (reference SAR Section 7.7.2). For a trip setting of 504 inches above vessel zero (144 inches above top of active fuel) and a 60-second valve closure time, the valves will be closed before perforation of the cladding occurs even for the maximum break; the setting is therefore adequate.

The low low reactor level instrumentation is set to trip when reactor water level is 444 inches above vessel zero (with top of active fuel defined as 360 inches above vessel zero, -59 inches is 84 inches above the top of active fuel). This trip initiates closure of Group 1 primary containment isolation valves, Ref. Section 7.7.2.2 SAR, and also activates the ECC subsystems, starts the emergency diesel generator and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious operation but low enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria.

indicative of a generic control rod drive problem and the reactor will be shutdown. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in Reference 6 can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would provide cause for suspecting a rod to be uncoupled and stuck. Restricting recoupling verifications to power levels above 20% provides assurance that a rod drop during a recoupling verification would not result in a rod drop accident more severe than analyzed.
2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.
3. Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the rod drop accident design limit of 280 cal/gam to be exceeded if they were to drop out of the core in the manner defined for the Rod Drop Accident.(5) These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM or a second qualified station employee. These sequences are developed to limit reactivity worths of control rods and together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 1.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2.1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

3.5 LIMITING CONDITION FOR OPERATION

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, at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5.1. 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.

4.5 SURVEILLANCE REQUIREMENT

I. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

heat generation rate even if fuel pellet densification is postulated. The power spike penalty is disccused in Reference 2 and assumes a linearly increasing variation in axial gaps between core bottom and top, and assumes with 95% confidence, that no more than one fuel rod exceeds the design LHGR due to power spiking.

K. Minimum Critical Power Ratio (MCPR)

The steady state values for MCPR specified in this Specification were selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state as well as uncertainties in the critical power correlation itself. These values also assure that operation will be such that the initial condition assumed for the LOCA analysis, plus 2% uncertainty is satisfied. For any of the special set of transients or disturbance caused by single operator error or single equipment malfunction, it is required that design analyses initialized at this steady state operating limit yield a MCPR of not less than that specified in Specification 1.1.A at any time during the transient assuming instrument trip settings given in Specification 2.1. For analysis of the thermal consequences of these transients, the value of MCPR stated in this specification for the limiting condition of operation bounds the initial value of MCPR assumed to exist prior to the initiation of the transients. This initial condition which is used in the transient analyses, will preclude violation of the MCPR fuel cladding integrity safety limit. Assumptions and methods used in calculating the required steady state MCPR limit for each reload cycle are documented in Reference 2. The results apply with increasing conservatism while operating with MCPRs greater than specified.

The most limiting transients with respect to MCPR are generally:

a) Rod withdrawal error
b) Load rejection or turbine trip without bypass
c) Loss of feedwater heater

Several factors influence which of these transients results in the largest reduction in critical power ratio such as the specific fuel loading, exposure, and fuel type. The current cycles reload licensing analyses specifies the limiting transient for a given exposure increment for each fuel type. The values specified as the Limiting Condition of Operation are conservatively chosen to bound the most restrictive over the entire cycle for each fuel type.

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the Specification. This assure that the MCPR will be maintained greater than that specified in Specification 1.1.A even in the event that the motor-generator set speed controller causes the scoop tube positioner for the fluid coupler to move to the maximum speed position.

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(2) "Generic Reload Fuel Application," NEDE-24011-P-A*

*Approved revision number at time reload fuel analyses are performed.

ENCLOSURE 1

Dresden Station Unit 2

Proposed Changes to Appendix A, Technical Specifications
To Facility Operating License DPR-19

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