DRAFT PROPOSED CHANGE #2

TECHNICAL SPECIFICATIONS AND BASES CHANGES

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 ~~With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:~~ (NOT USED)

~~THERMAL POWER shall be ≤ 25% RTP.~~

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ [1.07] for two recirculation loop operation or ≥ [1.08] for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL VIOLATIONS

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-----NOTE----- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 Item b or c requirements. ---------- Verify MFLPD is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter
SR 3.2.4.2	-----NOTE----- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 Item a requirements. ---------- Verify APRM setpoints or gains are adjusted for the calculated MFLPD.	12 hours

Add LCO 3.2.5 on next page

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Reactor Steam Dome Pressure and Core Flow

LCO 3.2.5 Reactor Steam Dome Pressure Shall Be ≥ [785 Psig]

AND

Core Flow Shall Be ≥ [10%] Of Rated.

APPLICABILITY: THERMAL POWER ≥ [25 %] RTP

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO Not met	A.1 Satisfy the Requirements of the LCO	2 hours
B. Required Action and Associated Completion Time not met.	B.1 Reduce THERMAL POWER to < [25%] RTP	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.5.1	Verify Reactor Steam Dome Pressure is ≥ [785 psig] AND Core Flow is ≥ [10 %] of Rated.	Once within 12 hours after ≥ [25%] RTP AND 24 hours thereafter

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

(NOT USED)

2.1.1.1~~a Fuel Cladding Integrity [General Electric Company (GE) Fuel]~~

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/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. With the design peaking factors, this corresponds to a THERMAL POWER > 50 % RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative.

2.1.1.1~~b Fuel Cladding Integrity [Advanced Nuclear Fuel Corporation (ANF) Fuel]~~

The use of the XN-3 correlation is valid for critical power calculations at pressures > 580 psig and bundle mass fluxes > 0.25 x 10^6 lb/hr-ft^2 (Ref. 3). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For the ANF 9x9 fuel design, the minimum bundle flow is > 30 x 10^3 lb/hr. For the ANF 8x8 fuel design, the minimum bundle flow is > 28 x 10^3 lb/hr. For all designs, the coolant minimum bundle flow and maximum flow area are

BASES

APPLICABLE SAFETY ANALYSES (continued)

such that the mass flux is always > 0.25×10^6 lb/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of > 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

2.1.1.2a MCPR [GE Fuel]

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.

2.1.1.2b MCPR [ANF Fuel]

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical

BASES

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. ~~SL 2.1.1.1 and~~ SL 2.1.1.2 ensure s that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs ~~2.1.1.1,~~ 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A (latest approved revision).
3. XN-NF524(A), Revision 1, November 1983.
4. 10 CFR 100.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared to FRTP or APRM gain or setpoints to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate gain or setpoint, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification than if MFLPD is less than or equal to fraction of rated power (FRP). When MFLPD is greater than FRP, more rapid changes in power distribution are typically expected.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23.
2. FSAR, Section [ ].
3. FSAR, Section [ ].

Add LCO B 3.2.5
on next page

B 3.2 POWER DISTRIBUTION SYSTEMS

B 3.2.5 REACTOR STEAM DOME PRESSURE AND CORE FLOW

BASES

BACKGROUND	The safety limit Minimum Critical Power Ratio (SLMCPR) is set such that 99.9% of the fuel rods will avoid the onset of transition boiling (OTB). The GEXL correlation is used by GE to calculate fuel bundle critical powers and is an NRC approved methodology. Similarly, the SPCB correlation is used for Framatone ANP (FANP) fuel. These methodologies are only approved, however, at high reactor pressures and core flows. During operations at normal rated conditions, and during anticipated operational occurrences (AOOs), fuel bundles are assured of not experiencing the onset of transition boiling (OTB) by adherence to the Operating Limit Minimum Critical Power Ratio (OLMCPR) provided in LCO 3.2.2. When operating at lower reactor pressures, or at low core flows, no approved methodology exists to assure that OTB will be avoided for any fuel bundle. Consequently, under low pressure or core flow conditions, reactor power must be reduced to a level which provides a large margin to the OTB.

APPLICABLE SAFETY ANALYSIS

GE Fuel

GE critical power correlations are applicable for all critical power calculations at pressures ≥ [785 psig] and core flows ≥ [10%] of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 1) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/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

APPLICABLE SAFETY ANALYSIS (Continued)

MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus a THERMAL POWER limit of [25%] RTP for reactor pressure < [785 psig] is conservative.

Framatone ANP (FANP) Fuel

The use of the SPCB correlation is valid for critical power calculations at pressures > 571.4 psig and bundle mass fluxes > .087 x 10^6 lb/hr-ft^2 (Ref 2). For operation at low pressures or low flows, fuel cladding integrity is protected by a limiting condition on core THERMAL POWER, with the following basis.

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For the FANP 10x10 and 9x9 designs, the minimum bundle flow is > 30 x 10^3 lb/hr. For the 8x8 design, the minimum bundle flow is > 28 x 10^3 lb/hr. For all designs, the coolant minimum bundle flow and maximum flow area are such that the mass flux is always > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle peaking factor of > 3.0. which is significantly higher than the expected peaking factor. Thus a THERMAL POWER limit of [25%] RTP for reactor pressures < [785 psig] is conservative.

LCO

The GEXL and SPCB correlations (used to perform the critical power correlations) are approved licensing models at greater than those pressures and flows listed in the BACKGROUND section. With reactor steam dome pressures less than [785 psig] or core flows less than [10%] of rated core flow the [GEXL/SPCB] correlation may not be valid. Therefore, the LCO

BASES

LCO
(Continued)

ensures that the reactor is not operated in a realm outside of the approved licensing basis for CPR calculations.

APPLICABILITY

Adherence to the LCO is required with THERMAL POWER ≥ [25%] of RTP.

At less than [25%] of RTP, the reactor is operating at minimum recirculation pump speed and the moderator void ratio is small. Therefore, remaining in an operational condition where the calculational basis for the CPR is valid is unnecessary due to the large inherent margins to OTB.

ACTIONS

A.1

If either the reactor steam dome pressure is less than [785 psig] or core flow is less than [10%] of rated with reactor power greater than [25%] of RTP, the reactor is operating outside analyzed conditions for OTB. Consequently, prompt action should be taken to restore operation to within the LCO limits. The 2 hour Completion Time is normally sufficient to restore reactor pressure or core flow to within limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the LCO out of specifications.

B.1

If reactor pressure and core flow cannot be restored to within the LCO limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. Therefore THERMAL POWER must be brought to less than [25%] of RTP within 4 hours. The Completion Time is reasonable, based on operating experience, to reduce the THERMAL POWER to less than [25%] of RTP in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

The reactor steam dome pressure and core flow limits are required to be verified once within 12 hours after exceeding 25% power to ensure that the reactor is operating within the [GEXL/SPCB] correlation basis. Afterwards, the 24 hour frequency is based on engineering judgment and that changes in reactor parameters such as pressure and core flow occur slowly during normal plant power changes, heat-ups or cooldowns.

REFERENCES

1. NEDE-24011-P-A. "General Electric Standard Application for Reactor Fuels" [revision specified in the COLR].

2. EMF-2209(P)(A) Revision 2, November, 2003.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Average Power Range Monitor

2.a. Average Power Range Monitor Neutron Flux - High, Setdown

The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP. For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, Setdown Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP ~~(SL 2.1.1.1)~~ LCO 3.2.5 when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux - High, Setdown with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This page is included for information only

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

<u>Main Steam Line Isolation</u>

<u>1.a. Reactor Vessel Water Level - Low Low Low, Level 1</u>

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level - Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level - Low Low Low, Level 1 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 100 limits.

This Function isolates the Group 1 valves.

<u>1.b. Main Steam Line Pressure – Low</u>

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. ~~In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)~~

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves.

<u>1.c. Main Steam Line Flow – High</u>

Main Steam Line Flow - High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow - High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 1). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.