B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SL

BASES

BACKGROUND

GDC 10 (Ref. 1) require that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of these SLs prevent overheating of the fuel and cladding, as well as possible cladding perforation, which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the peak linear heat rate (LHR) below the level that could cause fuel centerline temperature to reach the fuel rod melt temperature limit. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Protection System (PS), main steam safety valves (MSSVs), and main steam relief trains (MSRTs) prevents violation of the reactor core SLs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience centerline fuel melting.

The PS Limiting Trip Setpoints in LCO 3.3.1, Protection System (PS) in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, RCS Flow, AO, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

For DNBR limiting normal operation and AOO transients, three types of transients can be distinguished:

a. Type I: Transients from power for which the low DNBR protection channel of the PS is effective. These transients are protected by the Low DNBR Channel (LCO 3.3.1);

b. Type II: Transients for which the low DNBR protection is not effective. During these events protection is provided by another PS channel. This protection and the reserved DNBR margin included in LCO 3.2.3 “DNBR” make sure that the DNBR value at the initiation of the transient is greater than or equal to the value assumed in the safety analyses; and

c. Type III: Transients for which neither the “Low DNBR” protection channel nor the LCO 3.2.3 “DNBR” are sufficient.

BASES

APPLICABLE SAFETY ANALYSES (continued)

For type I transients:

The parameters that influence DNBR during the transient are taken into account in the low DNBR protection channel. The rate of change of the core boundary conditions is slow enough to be protected by the low DNBR protection channel. When the DNBR value calculated by the PS reaches the value of the trip setpoint, a reactor trip signal is issued. The probability that the actual DNBR value is higher than 1.0 is 95% with a 95% confidence level. Uncertainties and penalties are accounted for in the Low DNBR reactor trip setpoint in the incore static setpoint methodology.

For type II transients:

The transient is too rapid to be resolved by the low DNBR protection channel, so sufficient DNBR margin must be reserved at transient initiation to provide time for other plant trips to intercede and provide the needed protection. This is the purpose of LCO 3.2.2 “DNBR”. The DNBR LCO is established based on a combination of inherent system uncertainties and the results of the worst case transient DNBR degradation for those transients that are not protected by a Low DNBR trip.

During normal operation, if the LCO 3.2.2 is satisfied, the DNBR will remain higher than 1.0 with probability of 95% and a confidence level of 95% during a type II transient. Uncertainties and penalties are accounted for in the Low DNBR reactor trip setpoint in the incore static setpoint methodology.

For type III transients:

The “Low DNBR” protection channel and LCO 3.2.3 “DNBR” do not provide the required protection for the transient. For these transients, evaluations of the DNB performance during the event are performed using the approved sub-channel analysis code. Limiting conditions for core thermal power, core exit pressure, core inlet temperature, and core inlet mass flux are accounted for in these analyses. The minimum DNBR must remain greater than the DNBR design limit for the applicable critical heat flux correlation to make sure that the probability of preventing DNB is 95% with a 95% confidence level.

BASES

SAFETY LIMITS

SL 2.1.1.1 is the DNBR safety criterion of 1.0.

SL 2.1.1.2 makes sure that fuel centerline temperature remains below the fuel melt temperature of 4901ºF during normal operating conditions or design AOOs with an adjustment for burnup. An adjustment of 14ºF per 10,000 MWD/MTU has been established in Reference 3 for both UO_2 and UO_2:Gd_2O_3 fuel. Maintaining the dynamically adjusted peak LPD below the limit specified in the COLR ensures that fuel centerline melt will not occur during normal operating conditions or design AOOs.

For transients directly protected by the “High Linear Power Density” and “Low DNBR” protection channels, the LPD and DNBR PS setpoints will be set sufficiently higher (in the case of DNBR) or lower (in the case of LPD) with respect to the safety limit to account for the effects of various uncertainties. For DNBR, the correlation uncertainty is statistically taken into consideration when establishing the PS trip setpoint and the LCO setpoint.

APPLICABILITY

SLs 2.1.1.1 and 2.1.1.2 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The MSSVs, MSRTs, or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the PS functions are specified in LCO 3.3.1, "Protection System (PS)." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

BASES

SAFETY LIMIT VIOLATIONS	The following SL violation responses are applicable to the reactor core SLs. If SLs 2.1.1.1 or 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable. This ensures completion within 10 CFR 50.36(c)(1)(i)(a), which requires a shutdown when safety limits are violated. The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 10. 2. FSAR Section 7.2. 3. BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code,” Framatone ANP, September 2004.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2550 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety relief valves, main steam safety valves (MSSVs), main steam relief trains (MSRTs) and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressurizer safety relief valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Protection System setpoints (Ref. 5), together with the settings of the MSSVs and MSRTs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety relief valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of any of the following:

a. Turbine Bypass System,

b. Reactor Control System,

c. Pressurizer Level Control System, or

d. Pressurizer spray valve.

SAFETY LIMIT

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2803 psia.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

BASES

SAFETY LIMIT VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.

4. 10 CFR 100.

5. FSAR Section 7.2.