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CHAPTER B 2.0

SAFETY LIMITS (SLs)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires 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 requires that the minimum departure from nucleate boiling ratio (DNBR) of the limiting rod during Condition I and II events is greater than or equal to the DNBR correlation limits.

To meet this correlation limit design basis while accounting for uncertainties for Westinghouse fuel, for Revised Thermal Design Procedure (RTDP) analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation (WRB-2) predictions are combined statistically to obtain the overall DNBR uncertainty factor. This DNBR uncertainty factor is used to define the design limit DNBR, which corresponds to a 95% probability with 95% confidence that DNB will not occur on the limiting fuel rods during Condition I and II events. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their normal values. The design limit DNBR values are 1.21 and 1.22 for thimble and typical cells, respectively, for VANTAGE 5 fuel. In addition, margin has been maintained by meeting safety analysis DNBR limits above the design limit DNBR to offset known DNBR penalties and to provide DNBR margin for operating and design flexibility. Reference 3 discusses non-RTDP transients. These transients are analyzed using the WRB-2, W-3, ABB-NV, or WLOP DNB correlation, as applicable for the specific transient. The correlation limits for WRB-2, W-3, ABB-NV, and WLOP are 1.17, 1.30, 1.13, and 1.18, respectively.

For Framatome GAIA fuel, uncertainties are statistically applied to the boundary conditions analyzed rather than to the DNBR design limit. The DNBR safety limit for GAIA fuel assemblies is 1.12 for the ORFEO-GAIA Critical Heat Flux (CHF) correlation and 1.15 for the ORFEO-NMGRID CHF correlation with COBRA-FLX using the P-SCHEME Solver. The ORFEO-NMGRID CHF correlation DNBR safety limit is 1.18 in COBRA-FLX with the PV Solver. The DNBR safety limit is 1.12 for the ORFEO-GAIA CHF correlation and 1.15 for the ORFEO-NMGRID CHF correlation with XCOBRA-IIIC. Reference 5, 6, 7

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of

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fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. 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. Reference 4 further discusses the fuel centerline temperature design basis.

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 Reactor Trip System (RTS) and steam generator safety valves prevents violation of the reactor core SLs.

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 following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the limiting 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 Reactor Trip System Allowable Values in Table 3.3.1-1, 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, ΔI, 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.

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APPLICABLE SAFETY ANALYSES (CONTINUED)

Protection for these reactor core SLs is provided by the proper operation of the steam generator safety valves and the following automatic reactor trip functions:

a. High pressurizer pressure trip;

b. Low pressurizer pressure trip;

c. Low reactor coolant system flow;

d. Overtemperature ΔT trip;

e. Overpower ΔT trip; and

f. Power Range Neutron Flux trip.

The SLs represent a design requirement for establishing the RTS Allowable Values identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The reactor core safety limits figure provided in the COLR shows the loci of points of THERMAL POWER, pressurizer pressure, and average temperature below which the calculated DNBR is not less than the design limit DNBR values, the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation.

The reactor core SLs are established to preclude the violation of the following fuel design criteria:

a. There must be at least a 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. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core SLs are used to define the various RTS functions that the above criteria are satisfied during steady state operation, normal operating transients, and anticipated operational occurrences (AOOs). To ensure

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SAFETY LIMITS (CONTINUED)

that the RTS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature ΔT and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation.

Appropriate functioning of the RTS ensures that for variations in the THERMAL POWER, RCS pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients, and AOOs.

Reference 4 discusses the fuel temperature design basis. Figure 15.0-1 of Reference 2 depicts the protection provided by the Overpower ΔT reactor trip function against fuel centerline melting.

APPLICABILITY

SL 2.1.1 only applies 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 steam generator safety valves 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. Allowable Values for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

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, Chapter 15.
3. FSAR Section 4.4.1.1.
4. FSAR Section 4.4.1.2.

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REFERENCES (CONTINUED)

5. FS1-0026874, Revision 2.0, “CHF Correlation Applicability Limits.”

6. FS1-0050690, Revision 3.0, “ORFEO-GAIA and ORFEO-NMGRID Design Limit Validation for XCOBRA-IIIC.”

7. FS1-0058371, Revision 1.0, “Validation of the ORFEO-NMGRID CHF Correlation with the PV Solver.”

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 2485 psig. 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% (3110 psig) 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 50.67, "Accident Source Term" (Ref. 4).

APPLICABLE SAFETY ANALYSES

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

The RCS pressurizer safety 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

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APPLICABLE SAFETY ANALYSES (CONTINUED)

hence valve size requirements and lift settings, is a turbine trip without a direct reactor trip. Cases with and without pressurizer spray and PORVs are analyzed. Safety valves on the secondary side are assumed to open when the steam pressure reaches the safety valve settings. Main feedwater supply is lost at the time of turbine trip.

The Reactor Trip System Allowable Values in Table 3.3.1-1, together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization. The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

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

a. Pressurizer power operated relief valves (PORVs) (cases analyzed with and without);

b. Steam Generator Atmospheric Steam Dump valves;

c. Condenser Steam Dump valves;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valves (cases analyzed with and without).

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure.

The SL on maximum allowable RCS pressure is 2735 psig.

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.

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BASES (Continued)

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 50.67, "Accident Source Term," limits (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 50.67, "Accident Source Term."