

## 2.0 SAFETY LIMITS (SLs)

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### 2.1 SLs

#### 2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 5.41 MPaG or core flow < 10% rated core flow:

THERMAL POWER shall be  $\leq 25\%$  RTP.

2.1.1.2 With the reactor steam dome pressure  $\geq 5.41$  MPaG and core flow  $\geq 10\%$  rated core flow:

MCPR shall be  $\geq 1.07$ .

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  $\leq 9.13$  MPaG.

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### 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.

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## B 2.0 SAFETY LIMITS (SLs)

### B 2.1.1 Reactor Core SLs

#### BASES

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**BACKGROUND** GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling 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.

## BASES

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### 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, "SSLC Sensor Instrumentation"), in combination with all the 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.

#### 2.1.1.1 Fuel Cladding Integrity (General Electric Company (GE) Fuel)

GE critical power correlations are applicable for all critical power calculations at pressures  $\geq 5.41$  MPaG or core flows  $\geq 10\%$  of rated flow. For operation at low pressures and 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  $> 0.031$  MPa. Analyses (Ref. 2) show that with a bundle flow of  $12.7 \text{ m}^3/\text{h}$ , bundle pressure drop is nearly independent of bundle power and has a value of  $0.024$  MPa. Thus, the bundle flow with a  $0.031$  MPa driving head will be  $> 12.7 \text{ m}^3/\text{h}$ . Full scale ATLAS test data taken at pressures from  $0.098$  MPaA to  $5.5$  MPaA 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  $< 5.41$  MPaG is conservative.

#### 2.1.1.2 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

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APPLICABLE  
SAFETY  
ANALYSES  
(continued)

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 Ref. 2. Ref. 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.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

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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 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, thus maintaining a coolable geometry.

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APPLICABILITY

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

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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. 3). Therefore, it is required to insert all insertable control rods and restore compliance with the SL 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.

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 10.
  2. NEDE-24011-P-A, (latest approved revision).
  3. 10 CFR 100.
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## B 2.0 SAFETY LIMITS (SLs)

### B 2.1.2 Reactor Coolant System (RCS) Pressure SL

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**BACKGROUND** The SL on reactor steam dome pressure protects 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. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

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, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core is done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." 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 this occurred 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).

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**APPLICABLE SAFETY ANALYSES** The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure – High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, 1989 Edition, excluding Addenda (Ref. 5), which permits a maximum pressure transient of 110%, 9.48 MPaG, of design pressure 8.62 MPaG. The SL of 9.13 MPaG, as measured by the reactor steam dome pressure indicator, is equivalent to 9.48 MPaG at the lowest elevation of the RCS. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

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SAFETY LIMITS      The maximum transient pressure allowable in the RCS under the ASME Code, Section III, is 110% of design pressure. The SL on maximum allowable RCS pressure is established at 9.48 MPaG, which equates to 9.13 MPaG reactor steam dome pressure.

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APPLICABILITY      SL 2.1.2 applies in all MODES.

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SAFETY LIMIT VIOLATIONS      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," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action.

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- 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. Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
  4. 10 CFR 100.
  5. ASME, Boiler and Pressure Vessel Code, 1989 Edition, excluding Addenda.
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