ATTACHMENT 3

MARKED-UP TECHNICAL SPECIFICATIONS BASES
(FOR INFORMATION ONLY)

The following pages depict the changes proposed to the existing Technical Specification Bases. These pages are provided for information only with the final changes processed in accordance with the provisions of TS 5.5.14, "Technical Specifications (TS) Bases Control Program."

5 pages follow this cover sheet

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 fission products to the reactor coolant. Overheating of the fuel is

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BACKGROUND (CONTINUED)

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 Δ~~D~~I 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 Δ~~D~~T 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 notgenerating 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 and 4.7.4.1.2.

5. ANP-10341(P)(A), Revision 0, “The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations.”

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.”

Reviewer Note: FSAR Section 4.7.4 is a new FSAR Section to capture Framatome GAIA fuel-specific content.