Entergy

Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150
Tel 601 437 2800

May 8, 2001

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29
Technical Specification Bases Update to the NRC for Period April 27, 2001, through May 2, 2001

GNRO-2001/00037

Ladies and Gentlemen:

Pursuant to Grand Gulf Nuclear Station (GGNS) Technical Specification 5.5.11, Entergy Operations, Inc. hereby submits an update of all changes made to GGNS Technical Specification Bases since the last submittal (GNRO-2001/00035 letter dated April 27, 2001 to the NRC from GGNS). This submittal brings the Technical Bases up-to-date for the period April 27, through May 2, 2001. This update is consistent with update frequency listed in 10CFR50.71(e).

This letter does not contain any commitments.

Should you have any questions, please contact Rita Jackson at (601) 437-2149.

Yours truly,

Charles A. Bottemiller
Manager, Plant Licensing

RRJ/rrj
attachment: GGNS Technical Specification Bases Revised Pages
cc: (See Next Page)

cc:

Hoeg	T. L.	GGNS Senior Resident)	(w/a)
Levanway	D. E.	(Wise Carter)	(w/a)
Reynolds	N. S.		(w/a)
Smith	L. J.	(Wise Carter)	(w/a)
Thomas	H. L.		(w/o)

Mr. E. W. Merschoff (w/2) Regional Administrator U.S. Nuclear Regulatory Commission Region IV 611 Ryan Plaza Drive, Suite 400 Arlington, TX 76011
Mr. S. P. Sekerak, NRR/DLPM/PD IV-1 (w/2) **ATTN: ADDRESSEE ONLY** U.S. Nuclear Regulatory Commission One White Flint North, Mail Stop O7-D1 11555 Rockville Pike Rockville, MD 20852-2378

ATTACHMENT TO GNRO-2001/00037

GGNS Gulf Technical Specification Bases Revised Pages
for
Period April 27, 2001
Through
May 2, 2001

LDC#	BASES PAGES AFFECTED	TOPIC of CHANGE
00076	B 2.0-2, B 2.0-3, B 2.0-6,B 3.2-5, B 3.2-6, B 3.2-8	Fuel Reload for Cycle 12 Operation; Implementing Amendment No. 146 (GNRI-2001/00056)
01071	B 3.3-30a	Correct editorial errors in TRM Table 6.3.1-1 and Tech Spec Bases Figure B 3.3.1.1-1

BASES

BACKGROUND (continued)

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.

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 SL 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 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 SL.

2.1.1.1 Fuel Cladding Integrity

The use of the fuel vendor's critical power correlations are valid for critical power calculations at pressures $\geq$ 785 psig and core flows $\geq$ 10% of rated (Ref. 3, 5, and 6). 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:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flow will always be > 4.5 psi. Analyses 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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. Because of the design thermal hydraulic compatibility of the reload fuel designs with the cycle 1 fuel, this justification and the associated low pressure and low flow limits remain applicable for future cycles of cores containing these fuel designs.

2.1.1.2 MCPR

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 procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlation. Reference 2 describes the methodology used in determining the MCPR SL.

The calculated MCPR safety limit is reported to the customary three significant digits (i.e., X.XX); the MCPR operating limit is developed based on the calculated MCPR safety limit to ensure that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The fuel vendor's critical power correlations are based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlations, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the

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BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. ANF-524(P)(A), Revision 2, Supplements 1 and 2, November 1990.

3. EMF-2209(P)(A), Revision 1, July 2000.

4. 10 CFR 100.

5. Letter: CEXO-2000-00293, J. B. Lee (EOI) to K.V. Walters (SPC), “Grand Gulf Nuclear Station Unit 1 and Riverbend Station Unit 1, Reload Transition Data - GE11 Additive Constants”, July 25, 2000.

6. NEDE-24011-P-A, GESTAR-II.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Minimum Critical Power Ratio (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented *in the UFSAR, Chapters 4, 6, and 15, and References* 3, 4, and 5. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 3, 4, and 5). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 6) and the multi channel thermal hydraulic code (Ref. 2). $MCPR_f$ curves are provided based on the maximum credible flow runout transient for Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control.

Power dependent MCPR limits ($MCPR_p$) are determined by the three dimensional BWR simulator code and the one dimensional transient code (Ref. 2). The $MCPR_p$ limits are established for a set of exposure intervals. The limiting transients are analyzed at the limiting exposure for each interval. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limits are determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

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BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

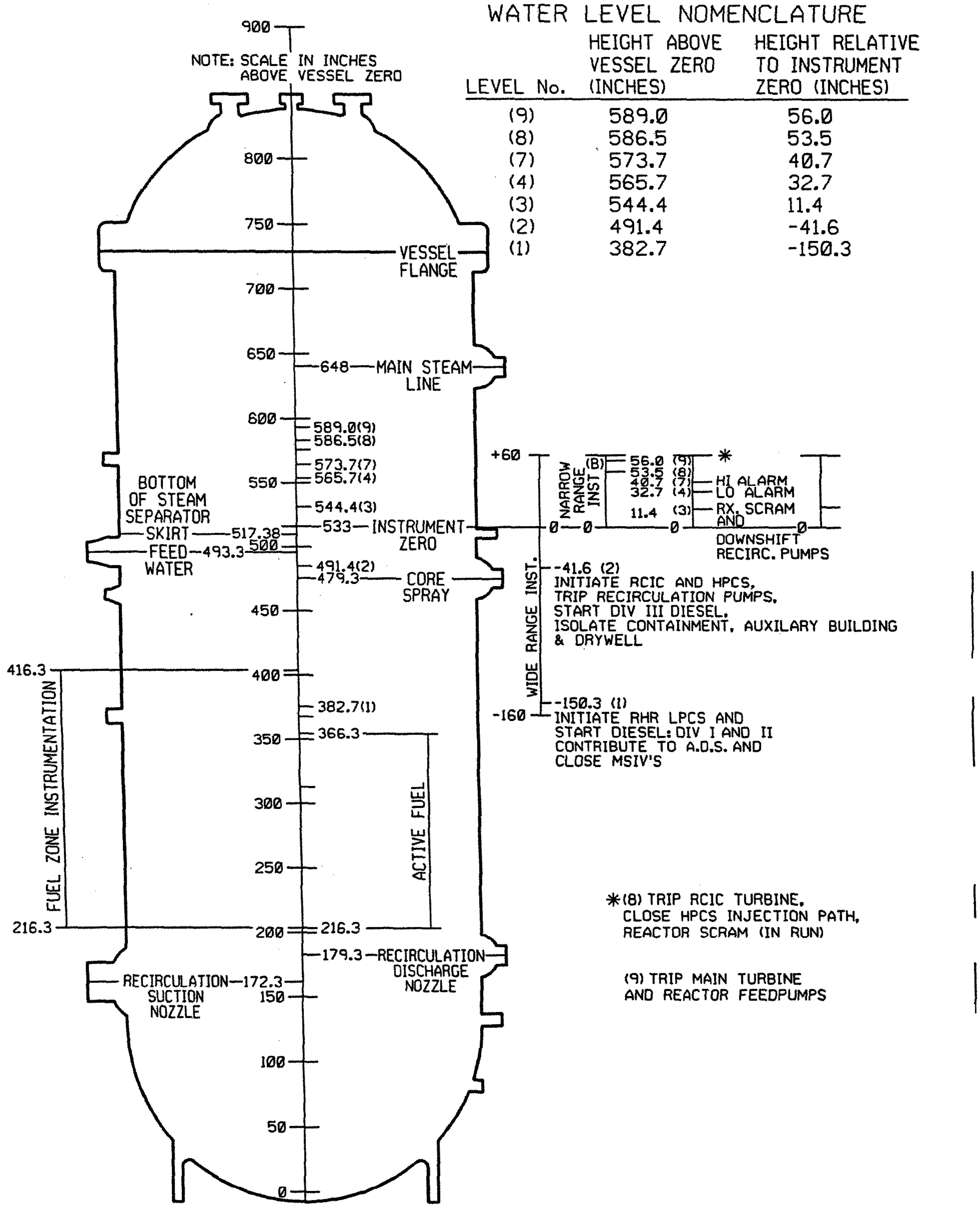
The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER reaches ≥ 25% RTP is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NUREG-0562, "Fuel Failures As A Consequence of Nucleate Boiling or Dry Out," June 1979.
2. XN-NF-80-19(P)(A) Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description", January 1987.
3. UFSAR, Chapter 15, Appendix 15B.
4. UFSAR, Chapter 15, Appendix 15C.
5. UFSAR, Chapter 15, Appendix 15D.
6. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis" March 1983 (As Supplemented).

WATER LEVEL NOMENCLATURE

LEVEL No.	HEIGHT ABOVE VESSEL ZERO (INCHES)	HEIGHT RELATIVE TO INSTRUMENT ZERO (INCHES)
(9)	589.0	56.0
(8)	586.5	53.5
(7)	573.7	40.7
(4)	565.7	32.7
(3)	544.4	11.4
(2)	491.4	-41.6
(1)	382.7	-150.3



BASES FIGURE B 3.3.1.1-1 REACTOR VESSEL WATER LEVEL