Entergy

Entergy
P.O. Box 756
Port Gibson, MS 39150
Tel 601 437 6409

William R. Brian
Vice President - Operations
Grand Gulf Nuclear Station

GNRO-2008/00006

February 14, 2008

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Supplement 3 to Amendment Request
Condensate Storage Tank Setpoint Change
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCES:

1. Letter GNRO-2007/00016 from W. R. Brian, Entergy Operations, Inc. to Document Control Desk, USNRC, "License Amendment Request Condensate Storage Tank Level-Low Setpoint Change," dated March 1, 2007 (TAC # MD 4675)

2. Letter GNRO-2007/00061 from W. R. Brian, Entergy Operations, Inc. to Document Control Desk, USNRC, "Supplement to Amendment Request Condensate Storage Tank Level-Low Setpoint Change," dated September 5, 2007.

3. Letter GNRO-2007/00068 from W. R. Brian, Entergy Operations, Inc. to Document Control Desk, USNRC, "Supplement 2 to Amendment Request Condensate Storage Tank Level-Low Setpoint Change," dated September 21, 2007.

Dear Sir or Madam:

By the Reference 1 letter above, Entergy Operations, Inc. (Entergy) proposed a change to the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specifications (TS) to incorporate the corrected allowable values in TS Tables 3.3.5.1-1 and 3.3.5.2-1.

The letter in Reference 2 above, documented responses to six questions resulting from calls held with the NRC staff to discuss the technical basis for the proposed TS change. The NRC staff had further requested copies of the surveillance procedures referenced in the supplemental letter, which were provided in Reference 3.

Several phone calls were held to discuss the responses and on December 19, 2007 the Staff requested that the provided information be docketed. A written explanation of the information presented is included in Attachment 1.

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NRR

There are no technical changes proposed. The original no significant hazards consideration included in Reference 1 is not affected by any information contained in the supplemental letter. There are no commitments contained in this letter.

If you have any questions or require additional information, please contact Bill Brice at 601-368-5076.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 14, 2008.

Sincerely,

W Brian

WRB/WBB

Attachment: Additional Information related to Changes to the CST Level-Low Setpoints

cc: Mr. Elmo E. Collins, Jr.
Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
61 1 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

U.S. Nuclear Regulatory Commission
ATTN: Mr. Bhalchandra Vaidya, NRR/DORL (w/2)
ATTN: ADDRESSEE ONLY
ATTN: U.S. Postal Delivery Address Only
Mail Stop OWFN/O-8G14
Washington, D.C. 20555-0001

Mr. Brian W. Amy, MD, MHA, MPH
Mississippi Department of Health
P. 0. Box 1700
Jackson, MS 39215-1700

NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Port Gibson, MS 39150

Attachment 1

To

GNRO-2008/00006

Additional Information Related to Changes to the CST Level-Low Setpoints

Additional Information Related to Changes to the CST Level-Low Setpoints

Letter GNRO-2007/00061 from W. R. Brian, Entergy Operations, Inc. to Document Control Desk, USNRC, "Supplement to Amendment Request Condensate Storage Tank Level-Low Setpoint Change," dated September 5, 2007 documented responses to six questions resulting from calls held with the NRC staff to discuss the technical basis for the proposed TS change. Several phone calls were held to discuss the responses and on 12-19-2007 the Staff requested that the provided information be docketed. A written explanation of the information presented in the phone calls is presented here. The discussions with the Staff focused on whether the CST low level transfer function was Safety Limit (SL) related. Entergy has concluded, based upon the below information, that the CST low level transfer function is not SL-related.

SL-Related LSSS and Proposed CST Level Setpoint Changes

Safety Limit Bases

Per TS B2.0, GDC 10 requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and Anticipated Operational Occurrences (AOO). The reactor core SL 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 in order to prevent elevated clad temperatures and resultant clad perforation. The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. The reactor water level SL is based on the concept that if the water level should drop below the top of the active irradiated fuel, the ability to remove decay heat is reduced. The SL on reactor steam dome pressure protects the Reactor Coolant System (RCS) against overpressurization.

10 CFR 50.36, "Technical Specifications":

Specifically Section 50.36(c)(1)(ii)(A) states: "Limiting Safety System Settings (LSSS) for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions" and requires that these LSSS are included in the TS. For variables on which a SL has been placed, the LSSS must be chosen to initiate automatic protective action to correct abnormal situations before the SL is exceeded. For example, the reactor water level setpoints credited in protecting the reactor water level SL, (i.e., Top of Active Fuel or TAF), following Anticipated Operational Occurrences (AOO) are chosen to ensure that makeup systems are initiated and water level is restored and maintained above TAF. 10 CFR 50.36(c)(1)(ii)(A) also contains requirements for a general class of LSSS. LSSS related to variables having significant safety functions but which do not protect SL. In these cases the settings must initiate automatic protective actions consistent with the design basis analyses for all postulated events including infrequent incidents and limiting faults (accidents). Event frequency classifications are described in UFSAR 15.0.3.1.

GGNS TS Change Request – CST Suction Transfer Setpoints

This TS change requested changes to the CST level allowable values (AV) that initiate the suction transfer for HPCS and RCIC from the CST to the suppression pool. This change has the potential to affect the response of HPCS and RCIC to postulated events including AOOs, infrequent incidents and limiting faults (i.e., accidents).

NRC Question on "SL Related" LSSS

The NRC questioned the relevance of TSTF-493 as discussed in RIS 2006-17 with regard to the proposed CST setpoints and the associated limiting trip setpoints (LSPs). Specifically the NRC requested that Entergy determine if the CST setpoints are "SL-Related" by providing a statement as to whether or not the relevant CST setpoint is a limiting safety system setting for a variable on which a SL has been placed as discussed in 10 CFR 50.36(c)(1)(ii)(A). As described above, 10 CFR 50.36(c)(1)(ii)(A) requires that the TS include LSSS for variables that have significant safety functions.

With respect to the "SL-Related" designation, RIS 2006-17 describes the LSP as "the limiting value to which the channel must be reset at the conclusion of periodic testing to ensure the SL will not be exceeded if a design basis event occurs before the next periodic surveillance or calibration." The use of the phrase "design basis event" is misleading in this situation because the events in question (those events where the SL is protected) do not include design basis accidents. SL are typically expected to be exceeded following a design basis accident. For example, in the design basis Loss of Coolant Accident (LOCA), MCPR decreases to well below 1.06 and water level decreases to well below TAF (see UFSAR Figures 6.3-12 and 6.3-14 for initial cycle). The acceptance criteria for design basis accidents are described in Section 15.0.3.1.3 of the USFAR "Unacceptable Results for Limiting Faults (Design Basis (Postulated) Accidents)", It states:

> The following are considered to be unacceptable safety results for limiting faults....design basis accidents:
>
> a. Radioactive material release which results in dose consequences that exceed the guideline values of 10 CFR 50.67
>
> b. Failure of fuel cladding which could cause changes in core geometry such that core cooling would be inhibited
>
> c. Nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes
>
> d. Containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required
>
> e. Radiation exposure to plant operations personnel in the main control room in excess of five Rem total effective dose equivalent in accordance with 10 CFR 50.67.

Since the CST suction transfer to the suppression pool is only credited in a design basis accident event and does not protect any reactor core safety limit, it not considered an SL related function. This is consistent with TS B2.0 and GDC 10.

The requested TS changes for the CST suction transfer to the suppression pool have the potential to impact long-term inventory source for the HPCS and RCIC systems following an AOO and therefore can potentially impact the TAF SL. A review of the AOO, where the RCIC and HPCS are discussed in the UFSAR, identified the Loss Of Feedwater (LOF) event as the limiting event relevant to RIS 2006-17. This AOO was determined to be the AOO that represents the greatest challenge to reactor inventory. If the CST level setpoints for the HPCS and RCIC and the associated suction transfer function are necessary to ensure that the applicable SL (e.g., TAF) is protected, these should be classified as SL-related LSSS in accordance with 50.36(c)(1)(ii)(A). The LOF event is only analyzed to demonstrate the acceptability of fuel thermal operating limits and, as such, is a short-term event that does not challenge the CST transfer function. The following discussion demonstrates that the suction transfer is not needed to protect the TAF SL following a LOF event based on a bounding analysis.

Loss of Feedwater Event Response

As discussed in the Amendment Supplement, a LOF event is described in GGNS UFSAR Section 15A.6.3.3 (Event 20) and in Section 15.2.7. As described in UFSAR 15.2.7.1.2, this event is a transient disturbance categorized as an incident of moderate frequency. Consequently, the plant response to this event requires that SL including fuel design limits are not exceeded per GDC 10. This event assumes that initial core cooling and reactor water level are maintained by either HPCS or RCIC. RCIC and HPCS are automatically actuated in this event by the Reactor Water Level (RWL) – Low Low, Level 2 signal. The CST low level suction swap function is not required for HPCS/RCIC actuation (actuated at RWL 2), reactor vessel injection (flow path established by normal lineup) or subsequent level restoration (sufficient inventory available – see discussion below). After level is restored by either the HPCS or RCIC systems, each system will function automatically to maintain reactor water level between reactor level 2 and reactor level 8. If the additional water source contribution from the operating control rod drive pump is ignored, the makeup requirements are essentially the same as those following a Station Blackout (SBO) event (i.e., boil-off from decay heat). As discussed in the original submittal, CST inventory is sufficient to accommodate the inventory requirements for the SBO event for the coping period of 4 hours. The SBO analysis assumes 115,278 gallons are injected into the reactor vessel during this period. Because this volume is less than the available inventory in the CST, assuming an initial water level at the low level alarm (22 ft), neither the RCIC or HPCS systems will automatically attempt to transfer suction to the suppression pool until well beyond four hours – even with the revised setpoints. Therefore, the CST level setpoint is based on ensuring this function is performed following all postulated events and is not needed to protect the TAF SL following the LOF event.

Conclusion

As discussed in RIS 2006-17, 10 CFR 50.36(c)(1)(ii)(A) also contain requirements for a general class of LSSS that do not protect a SL. Since SL avoidance is not applicable to infrequent events or accidents, the postulated events relevant to the "SL-related" designation only include normal operation and AOO. Safety limits are exceeded in design basis accidents.

Since the CST suction transfer to the suppression pool is only credited in a design basis accident event, and does not protect any reactor core SL, it is not an SL-related function. This is consistent with TS B2.0 and GDC 10.