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GNRO-2007/00061

September 5, 2007

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

SUBJECT: Supplement to Amendment Request
Changes to the Condensate Storage Tank Level-Low Setpoints
Grand Gulf Nuclear Station, Unit 1 (GGNS)
Docket No. 50-416
License No. NPF-29

REFERENCE: 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)

Dear Sir or Madam:

By the above referenced letter, 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.

Entergy and members of your staff held calls to discuss the technical basis for the proposed TS change. As a result of the calls, six questions were determined to need formal response: four of the questions from the Instrumentation and Controls Branch and two questions from the Reactor Systems Branch. Entergy's response is contained in Attachment 1.

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

If you have any questions or require additional information, please contact Matt Crawford at 601-437-2334.

ADD
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 5, 2007.

Sincerely,



WRB/MLC

Attachments:

1. Response to Request For Additional Information
2. Sketch of HPCS/RCIC CST Suction Piping Diagram
3. GGNS Calculation JC-Q1E22-N654-1, Rev. 3
4. GGNS Calculation JC-Q1E51-N635-1, Rev. 1
5. GGNS Calculation M6.7.013, Rev. 1

cc: Dr. Bruce S. Mallett
Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
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Arlington, TX 76011-4005

U.S. Nuclear Regulatory Commission
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NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Port Gibson, MS 39150

Attachment 1

To

GNRO-2007/00061

Response to Request for Additional Information

**Response to Request for Additional Information Related to
Revising the Condensate Storage Tank Level-Low Setpoints**

1. Please provide the physical layout diagrams of the condensate storage tank (CST) and the non-safety related HPCS/RCIC suction piping associated with the new calculated values. Also, provide the calculations used to determine the new values.

Response:

A simplified diagram of the piping arrangement labeled "HPCS/RCIC CST SUCTION PIPING DIAGRAM" is provided as Attachment 2. Copies of the setpoint calculations JC-Q1E22-N654-1, Rev 3 (Attachment 3) and JC-Q1E51-N635-1, Rev 1 (Attachment 4) are also included.

2. Please provide the calculations, including the physical dimensions of the CST, used to determine that for a station blackout the 115,278 gallons required to cope with a four hour SBO event is still satisfied with the new setpoint value.

Response:

A copy of calculation M6.7.013, Rev 1 is provided as Attachment 5. Per this calculation, a CST level of 18.9 ft (standpipe elevation) ensures a usable reserve volume of 143,000 gallons at 800 gpm with the new 4 ft setpoint.

3. Setpoint Calculation Methodology: Provide documentation (including sample calculations) of the methodology used for establishing the limiting setpoint (or NSP) and the limiting acceptable values for the As-Found and As-Left setpoints as measured in periodic surveillance testing described below. Indicate the related Analytical Limits and other limiting design values (and the sources of these values) for each setpoint.

Response:

The Nominal Trip Setpoint (NTSP) values and the proposed tech spec allowable values are derived in calculations JC-Q1E22-N654-1 (Attachment 3) and JC-Q1E51-N635-1 9 (Attachment 4). The GE NEDC 31336P-A setpoint methodology is utilized. The As-Found and As-Left tolerances of the setpoints specified in the plant surveillance procedures 06-IC-1E22-Q-0002, 06-IC-1E51-Q-0002, 06-IC-1E22-R-0002 and 06-IC-1E51-R-0002 is 0.25% span per JC-Q1E22-N654-1 and JC-Q1E51-N635-1. The associated analytical limits are also discussed in the calculations.

4. Safety Limit (SL)-Related Determination: Provide a statement as to whether or not the setpoint is a limiting safety system setting for a variable on which a safety limit (SL) has been placed as discussed in 10 CFR 50.36(c)(1)(ii)(A). Such setpoints are described as "SL-Related" in the discussions that follow. In accordance with 10 CFR 50.36(c)(1)(ii)(A), the following guidance is provided for identifying a list of functions

to be included in the subset of LSSSs specified for variables on which SLs have been placed as defined in Standard Technical Specifications (STS) Sections 2.1.1, Reactor Core SLs and 2.1.2, Reactor Coolant System Pressure SLs. This subset includes automatic protective devices in TSs for specified variables on which SLs have been placed that: (1) initiate a reactor trip; or (2) actuate safety systems. As such these variables provide protection against violating reactor core safety limits, or reactor coolant system pressure boundary safety limits.

Examples of instrument functions that might have LSSSs included in this subset in accordance with the plant-specific licensing basis, are pressurizer pressure reactor trip (pressurized water reactors), rod block monitor withdrawal blocks (boiling water reactors), feedwater and main turbine high water level trip (boiling water reactors), and end of cycle recirculation pump trip (boiling water reactors). For each setpoint, or related group of setpoints, that you determined not to be SL-Related, explain the basis for this determination.

Response:

The affected setpoints are not "SL-Related" since they do not meet the specified criteria. TS 2.1.1, Reactor Core Safety Limits, requires the reactor vessel water level to be greater than the top of active irradiated fuel (TAF). A Loss of Feedwater Event described in GGNS Updated FSAR section 15A.6.3.3, Event 20 and in section 15.2.7 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 -Low Low, Level 2 signal. The CST low level signal is not required for HPCS/RCIC actuation or reactor vessel injection. The CST low level signal along with the associated automatic transfer function ensures a continuous inventory supply for the HPCS and RCIC systems after the reactor vessel is refilled as part of its long-term cooling function. The CST level setpoint is based on ensuring the suction source transfer is performed and is not related to protecting the TAF or any other safety limit. Therefore, this setpoint is not SL-related.

5. For setpoints that *are* determined to be SL-Related: The NRC letter to the NEI SMTF dated September 7, 2005 (ML052500004), describes Setpoint-Related TS (SRTS) that are acceptable to the NRC for instrument settings associated with SL-related setpoints. Specifically: Part "A" of the Enclosure to the letter provides LCO notes to be added to the TS, and Part "B" includes a check list of the information to be provided in the TS Bases related to the proposed TS changes.
 - a. Describe whether and how you plan to implement the SRTS suggested in the September 7 letter. If you do not plan to adopt the suggested SRTS, then explain how you will ensure compliance with 10 CFR 50.36 by addressing items 3b and 3c, below.
 - b. As-Found Setpoint evaluation: Describe how surveillance test results and associated TS limits are used to establish operability of the safety system. Show that this evaluation is consistent with the assumptions and results of the setpoint calculation methodology. Discuss the plant corrective action processes (including plant procedures) for restoring

channels to operable status when channels are determined to be "inoperable" or "operable but degraded." If the criteria for determining operability of the instrument being tested are located in a document other than the TS (e.g. plant test procedure) explain how the requirements of 10 CFR 50.36 are met.

- c. As-Left Setpoint control: Describe the controls employed to ensure that the instrument setpoint is, upon completion of surveillance testing, consistent with the assumptions of the associated analyses. If the controls are located in a document other than the TS (e.g. plant test procedure) explain how the requirements of 10 CFR 50.36 are met.

Response:

The affected setpoints are not "SL-Related"; therefore, this question is Not Applicable.

6. For setpoints that are *not* determined to be SL-related: Describe the measures to be taken to ensure that the associated instrument channel is capable of performing its specified safety functions in accordance with applicable design requirements and associated analyses. Include in your discussion information on the controls you employ to ensure that the as left trip setting after completion of periodic surveillance is consistent with your setpoint methodology. Also, discuss the plant corrective action processes (including plant procedures) for restoring channels to operable status when channels are determined to be "inoperable" or "operable but degraded." If the controls are located in a document other than the TS (e.g., plant test procedure), describe how it is ensured that the controls will be implemented.

Response:

The trip unit setpoints are checked approximately every 92 days per surveillance procedures 06-IC-1E22-Q-0002, 06-IC-1E51-Q-0002. The transmitter calibration and trip unit calibration is checked approximately every 18 months per surveillance procedures 06-IC-1E22-R-0002 and 06-IC-1E51-R-0002. The As-Found and As-Left tolerance of the transmitter and trip units is 0.25% span per JC-Q1E22-N654-1(Attachment 4) and JC-Q1E51-N635-1 (Attachment 5). If one of the transmitters or trip units is found out of calibration, it will be recalibrated per the surveillance procedure. In addition, a condition report (CR) will be initiated if the Technical Specification allowable value is exceeded. Actions taken with an inoperable channel are in accordance with the Technical Specification.

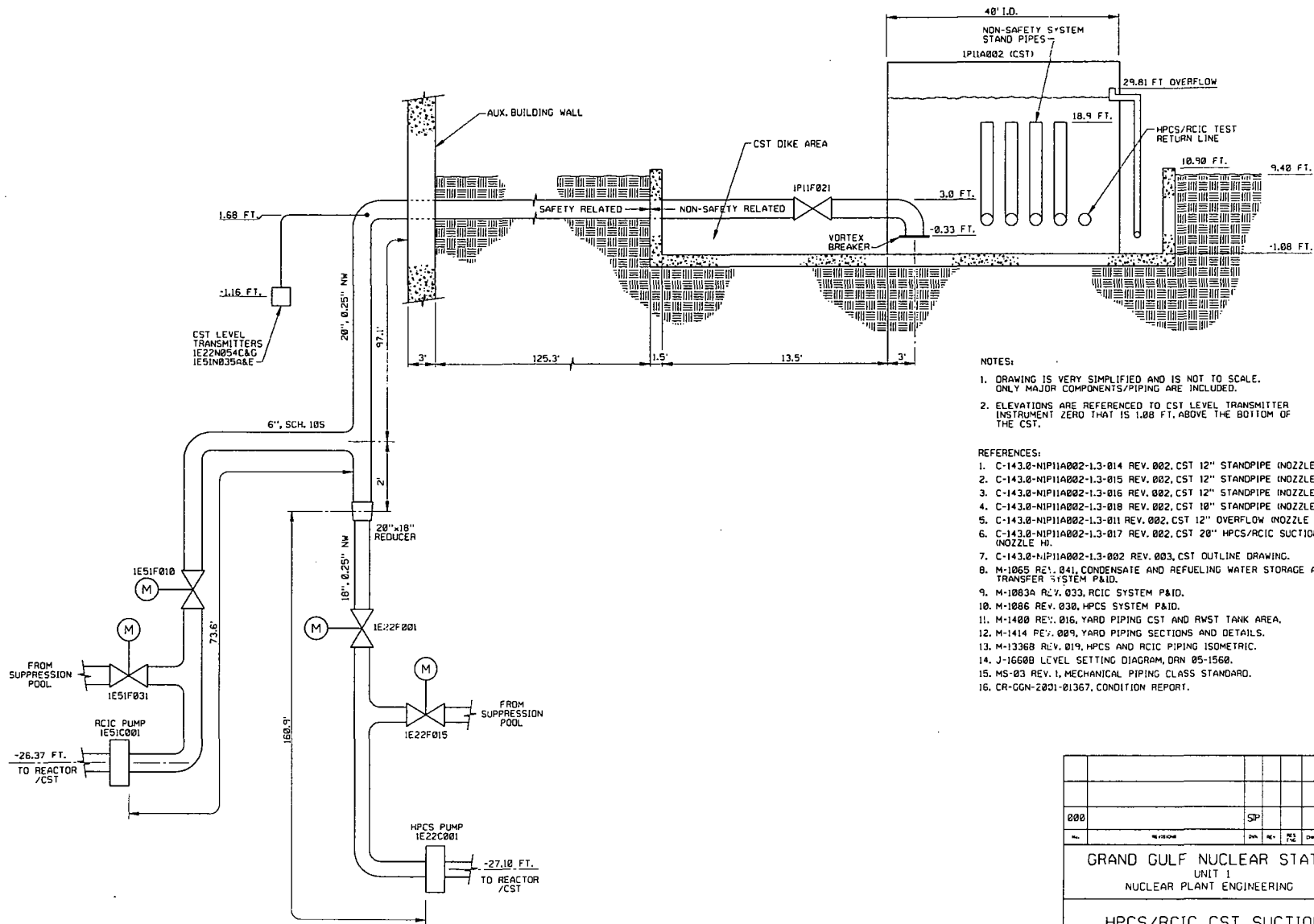
Attachment 2

Attachment 2

To

GNRO-2007/00061

Sketch of HPCS/RCIC CST Suction Piping Diagram



NOTES:

1. DRAWING IS VERY SIMPLIFIED AND IS NOT TO SCALE. ONLY MAJOR COMPONENTS/PIPING ARE INCLUDED.
2. ELEVATIONS ARE REFERENCED TO CST LEVEL TRANSMITTER INSTRUMENT ZERO THAT IS 1.08 FT. ABOVE THE BOTTOM OF THE CST.

- REFERENCES:
1. C-143.0-NIP11A002-1.3-014 REV. 002, CST 12" STANDPIPE (NOZZLE E).
 2. C-143.0-NIP11A002-1.3-015 REV. 002, CST 12" STANDPIPE (NOZZLE F).
 3. C-143.0-NIP11A002-1.3-016 REV. 002, CST 12" STANDPIPE (NOZZLE G).
 4. C-143.0-NIP11A002-1.3-018 REV. 002, CST 10" STANDPIPE (NOZZLE J).
 5. C-143.0-NIP11A002-1.3-011 REV. 002, CST 12" OVERFLOW (NOZZLE L).
 6. C-143.0-NIP11A002-1.3-017 REV. 002, CST 20" HPCS/RCIC SUCTION (NOZZLE H).
 7. C-143.0-NIP11A002-1.3-002 REV. 003, CST OUTLINE DRAWING.
 8. M-1065 REV. 041, CONDENSATE AND REFUELING WATER STORAGE AND TRANSFER SYSTEM P&ID.
 9. M-1083A REV. 033, RCIC SYSTEM P&ID.
 10. M-1086 REV. 030, HPCS SYSTEM P&ID.
 11. M-1400 REV. 016, YARD PIPING CST AND RWST TANK AREA.
 12. M-1414 REV. 009, YARD PIPING SECTIONS AND DETAILS.
 13. M-13368 REV. 019, HPCS AND RCIC PIPING ISOMETRIC.
 14. J-16600 LEVEL SETTING DIAGRAM, DSN 05-1560.
 15. MS-03 REV. 1, MECHANICAL PIPING CLASS STANDARD.
 16. CR-CGN-2001-01367, CONDITION REPORT.

000	SP				
GRAND GULF NUCLEAR STATION UNIT 1 NUCLEAR PLANT ENGINEERING					
HPCS/RCIC CST SUCTION PIPING DIAGRAM					
SCALE:	NONE	SKETCH No.		REV.	000