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GNRO-2008/00078

January 19, 2009

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

SUBJECT: Response to Basis for Denial of Condensate Storage Tank Setpoint Change letter dated November 5, 2008
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 from NRC "Grand Gulf Nuclear Station, Unit 1 – Staff Evaluation and Basis for Denial of Proposed Technical Specification Change Related to the Condensate Storage Tank Level-Low Setpoint Changes (TAC No. MD4675)"

Dear Sir or Madam:

By Reference 1, Entergy Operations, Inc. (Entergy) proposed a change to the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specifications (TS) to correct allowable values in TS Tables 3.3.5.1-1 and 3.3.5.2-1. By Reference 2, the NRC Staff provided an evaluation and basis for denial of the proposed change. The staff's conclusion was that the change for the High Pressure Core Spray System (HPCS) instrument setpoint is Safety Limit (SL) - related and that additional requirements must be added to the TS to address the concerns of RIS 2006-17 (e.g., new specific footnotes). Reference 2 requested Entergy's review and comments on the staff's evaluation. Entergy has completed the review and provided comments in Attachment 1.

Entergy contends that the requested change should not be denied based upon the NRC staff positions presented in RIS 2006-17. Entergy believes that these positions may constitute new requirements that need further regulatory review and processing prior to being imposed upon licensees. The requested change is needed to maintain the current GGNS licensing basis conservative with respect to the design of the plant. Denial of the request would leave the GGNS TS requirement to be non-conservative. Entergy believes that safety and the stability of the regulatory process would best be served by promptly approving the amendment request while allowing industry activities to reach a generic solution of the issues discussed in RIS 2006-17.

Entergy is monitoring the efforts of the Technical Specification Task Force and NRC to finalize the details and scope of the changes needed to resolve the instrument setpoint issue discussed in RIS-2006-17. We plan on supporting any NRC and industry solutions when finalized.

The original no significant hazards consideration included in Reference 1 is not affected by any information contained in the supplemental letters.

If you have any questions or require additional information, please contact Bryan Ford at 601-368-5516.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 19, 2009.

Sincerely,

M. A Krupa

MAK/WBB

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

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Attachment 1

To

GNRO-2008/00078

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

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

1.0 DESCRIPTION

By letter dated November 5, 2008 the NRC staff provided to Grand Gulf Nuclear Station, Unit 1, (GGNS) a Staff Evaluation and Basis for Denial of Proposed Technical Specification change related to the Condensate Storage Tank (CST) Level-Low setpoint changes (TAC No. MD4675). This was in response to the application dated March 1, 2007 (GNRO-2007/00016) as supplemented by letters dated September 5, 2007 (GNRO-2007-00061) September 21, 2007 (GNRO-00068), and February 14, 2008 (GNRO-2008/00006). The proposal was to change the allowable values for the CST level-low setpoints in the GGNS Technical Specifications (TS).

The proposed change would revise the Operating License of GGNS to provide more conservative low water level suction swap setpoint allowable values for the High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) systems from the Condensate Storage Tank (CST) to the Suppression Pool.

2.0 BACKGROUND

The CST is designed to store condensate for the RCIC and the HPCS systems and to maintain the level of condensate in the condenser hotwell, and provide condensate to other plant systems, where required. The CST is a stainless steel storage tank with a capacity of 300,000 gallons. The system has no safety related function.

The CST level is normally maintained above 25 feet and has a low level alarm at 22 feet. The CST level-low alarm warns of a low level in the CST which in turn indicates the potential unavailability of an adequate supply of makeup water. The CST also provides makeup water to several other systems. The CST reserves a volume specifically for HPCS and RCIC. This is accomplished by the use of standpipes inside the CST that ensures that the non-safety systems cannot draw the CST level below 18.9 feet. 18.9 feet equates to approximately 180,000 gallons of water in the tank.

The accident analyses assume that the Suppression Pool is the suction source for HPCS while RCIC does not have a required post accident function. Normally the suction valves between HPCS/RCIC and the CST are open and, upon receiving a HPCS/RCIC initiation signal, water for injection is taken from the CST. If the level in the CST drops below a predetermined level (i.e., the low level suction swap setpoint), the Suppression Pool suction valve would open and then the CST suction valve would close. This ensures an adequate supply of water for the HPCS/RCIC systems during accident conditions. Although the CST is the preferred and normal source of water for these systems, it is not a seismic Category 1 structure, and is not credited in an accident as a HPCS suction source. Therefore, an automatic safety-grade suction switchover to the Suppression Pool is provided. This suction switchover function is required for an operable HPCS/RCIC system in accordance with the TS.

The CST level-low signals are initiated through two transmitters either of which can affect an automatic suction swap. These transmitters are not connected to the CST, but instead are connected to the HPCS/RCIC suction line inside the Auxiliary Building and are safety related. It was discovered that these transmitters may not have been capable of providing the CST low level trip that would transfer the HPCS/RCIC suction from the CST to the Suppression Pool under all conditions. This is because the transmitters have an uncorrected static head between the tank and the safety related portion of the suction pipe. This static head in the safety related tubing is normally offset by the inverted HPCS/RCIC suction nozzle inside the CST. This would not be the case if a seismic or other event were to occur that resulted in draining of the CST. Resetting of the low level suction swap setpoints resolved this issue by raising the minimum water level in the CST above the level of the HPCS/RCIC suction piping. The prior value would not have assured a suction swap-over in the event of a seismic or other event that would have caused a failure in the non-seismic portion of the system. With this change, in the event of a failure of the HPCS/ RCIC piping, an automatic swap to the Suppression Pool will occur.

3.0 Analysis of the Basis for Denial

In reviewing Entergy's request, the NRC staff concluded that the HPCS Allowable Value (AV) is safety limit related. The conclusion was based on several main points. Entergy is providing comments on some of those points below.

In the referenced letter, the NRC Staff reached the following conclusion:

> "The NRC staff reviewed the licensee's rationale and finds that the analysis is not complete. The NRC staff finds that the CST level-low setpoint would not be required for LOF, as suggested by the licensee, only if the non-safety-grade CST is available. The licensee states that high pressure makeup systems will not attempt to transfer for 4 hours. However, this statement implies that the non-safety-grade CST is credited for the duration. In accordance with 10 CFR 50.2 and 50.49, AOOs analyses should credit only safety-grade systems for event mitigation. The NRC staff believes that the licensee's rationale, which claims the transfer is not necessary, implicitly credits the CST level-low function to mitigate the LOF. If the non-safety grade CST were not credited, then the CST level-low setpoint would be required to ensure the availability of safety-grade suction source.
>
> Based on the above, for the use of HPCS in accordance with 10 CFR 50.2 and 50.49, the NRC staff concludes that the automatic transfer from the CST to the suppression pool is required for the HPCS system to comply with the plant design basis. Therefore, on this basis, the NRC staff has concluded that the HPCS AV for this automatic transfer is SL-related. This is consistent with the statement in UFSAR Section 6.3.1.1.1.d that the HPCS system is provided for maintaining the water level above the top of the core. "

NRC Position 1

The loss-of-feedwater flow Anticipated Operational Occurrence (AOO) analysis cannot credit availability of the CST (i.e., the CST is implicitly assumed to fail).

Response:

The HPCS swap over from the CST to the Suppression Pool is a safety related function. It is a safety related function to support HPCS Operability during a Small Break Loss of Coolant Accident (SBLOCA). The CST to Suppression Pool swap over would also be required to function to mitigate the loss of feedwater AOO if the CST were assumed to fail. However, the assumption of a failure of the passive components of the CST during a loss of feedwater transient would be beyond the GGNS licensing basis for AOOs.

The GGNS UFSAR defines incidents of moderate frequency (Anticipated Operational Occurrences or AOOs) as those incidents that may occur during a calendar year to once in 20 years. Thus incidents or events occurring with a frequency of less than 5E-02/yr are not considered an AOO. The identified concern involves a loss of feedwater initiating event followed by the loss or failure of the CST which would require the realignment of HPCS suction from the CST to the Suppression Pool. A loss of feedwater event results in an automatic reactor scram and subsequent initiation of HPCS in the limiting analysis. HPCS is then assumed to operate to provide makeup to the Reactor Coolant System (RCS) until the reactor is sufficiently depressurized such that the Residual Heat Removal System (RHR) can be placed in service in the Shutdown Cooling mode of operation. This time frame is conservatively assumed to be the 6 hours immediately following the initiating event (loss of feedwater). Sufficient water is reserved by the passive design of the CST (i.e., the standpipe in the CST) to ensure that minimum volume of water to support RCS makeup is available.

Failure of the CST during a LOF event is not a moderate frequency event and would therefore be evaluated against different criteria which would not include safety limit protection. The following discussion demonstrates that the probability of these events occurring concurrently is much less than that necessary to require an AOO analysis. The time frame for failure of the CST is conservatively assumed to be within 6 hours immediately following the initiating event (LOFW) in the below evaluation.

Passive Failure of the CST

U. S. Department of Energy document EGG-SSRE-9639, "Component External Leakage and Rupture Frequency Estimates", provides a value of 1.0E-08/hr for the rate of tank leakage and a value of 4.0E-10/hr for the rate of tank rupture. If the higher failure rate is used to determine the probability of failure of the CST within 6 hours, the failure probability is 6.0E-08. Thus, assuming the loss of feedwater occurs once per year, the frequency of the combination of the two events is 1/yr X 6.0E-08 or 6.0E-08/yr. This frequency is well below the criteria for inclusion in the AOO category.

Failure of CST due to Tornado

GGNS calculation CC-Q1111-94004, "Probabilistic Evaluation of Tornado Missile Strike for IPEEE Study," provides an analysis of the likelihood of a tornado missile strike on several specific targets at GGNS. The CST is not identified as one of the targets. However, as part of the calculation the frequency of a tornado strike in Region I, defined in Regulatory Guide 1.72, "Design Basis Tornado for Nuclear Power Plants," was determined for all tornado intensities. GGNS is located in Region 1 and the frequency of a tornado strike in Region 1 can be conservatively utilized in lieu of the probability of a tornado missile striking the CST. This is conservative since the probability of a direct missile strike and the probability of CST failure given a strike are not considered. From CC-Q1111-94004, the total frequency of a tornado strike is 4.25E-04/yr. Therefore, the probability of a tornado strike in a 6 hour period is 3.08E-07 (4.25E-04/yr X 6 hrs / 8760 hrs/yr). Thus, assuming the loss of feedwater occurs once per year, the frequency of the combination of the two events is 1/yr X 3.08E-07 or 3.08E-07/yr. This frequency is well below the criteria for inclusion in the AOO category.

Failure of CST due to Seismic Event

According to the UFSAR, the GGNS ground peak acceleration for the safe shutdown earthquake (SSE) is 0.15g. The ground peak acceleration for the operating basis earthquake is 0.075g. The CST is not specifically designed for either of these earthquakes but is instead designed to Uniform Building Code (UBC) Zone 1 requirements. The determination of the equivalent ground peak acceleration for these requirements is not straight forward. Therefore, instead of calculating the probability of a specific level of earthquake which would fail the CST in the 6 hours following an initiator, the required frequency that would cause the combination of an initiating event followed by the seismic event to be considered an AOO is calculated. As discussed earlier, the incidents of moderate frequency (AOOs) are those incidents that may occur during a calendar year to once in 20 years (i.e., 5E-02/yr). Thus incidents or events occurring with a lower frequency are not considered AOOs.

Given that the initiator (loss of feedwater) has occurred, the probability of an earthquake in the following 6 hours must be 5E-02 in order for the frequency of combined events to be considered an AOO. The annual frequency of an earthquake whose probability for a 6 hour period is 5E-02 is 73/yr (5E-02 X 8760 hrs/yr/6 hrs). Earthquakes with a frequency of 73/yr would be associated with very low ground accelerations and therefore are not expected to lead to a failure of the CST. As data from NUREG-1488, "Revised Livermore Seismic Hazard Estimate for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," indicates, the annual frequency of an earthquake is inversely related to its severity. For example see the GGNS data in the following table.

Acceleration, g	Acceleration, cm/sec/sec	Mean Frequency
0.051	50	3.306E-04
0.075 (OBE)	73.5	~1.765E-04[1]
0.15 (SSE)	147	~5.513E-05[2]

1. Actually associated with 75 cm/sec/sec
2. Actually associated with 150 cm/sec/sec

The data in this table would indicate that the seismic ground peak acceleration with a frequency of 73/yr would be very low and therefore, within the capability of the CST as well as almost all commercial or domestic structures. It should also be noted that the non-safety related pipe connected to the CST has a greater ability to withstand seismic activity than the CST itself.

Based on the above, it is concluded that the frequency of a loss of feedwater followed by a seismic event within 6 hours which would fail the CST is well below that necessary to classify the combination of events into the AOO category. As a result, the HPCS swap over from the CST to the Suppression pool as a suction source is not required to be credited during the loss of feedwater AOO.

NRC Position 2

In Section 4.2 of the denial evaluation, the Staff states that "In accordance with 10 CFR 50.2 and 50.49, AOOs analyses should credit only safety-grade systems for event mitigation."

Response:

The conclusion that only safety-grade systems can be credited for event mitigation is not consistent with previous Staff positions or the GGNS licensing basis. A discussion of the regulations, NRC guidance and the GGNS licensing basis concerning credit for non-safety grade equipment in plant analyses is provided below.

10 CFR 50.49

This regulation is not applicable to the subject request. 10 CFR 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants, only applies to electrical equipment and; therefore, does not apply to the CST or associated piping.

10 CFR 50.2

10 CFR 50.2 defines the three areas where safety-related structures, systems and components are required to remain functional in response to design basis events:

(1) The integrity of the reactor coolant pressure boundary

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in Sec. 50.34(a)(1) or Sec. 100.11 of this chapter, as applicable.

None of these criteria apply to the LOF AOO. Although utilized in response to the LOF AOO, neither the integrity of the reactor coolant pressure boundary or shutdown capability is challenged. The CST transfer function does not play a role in any of these areas and each of these areas is fully protected by safety-related SSCs. For a LOF event, the unit can be maintained in a safe shutdown condition as described in Updated FSAR section 18.1.30.15.

This FSAR section provides a response to TMI-related item II.K.3.44 identified in NUREG-0737 regarding the issue of adequate core cooling for transients with a single failure. A BWR Owner's Group generic evaluation representative of GGNS determined that during a LOF transient combined with a failure of HPCS and a stuck open relief valve, the core remained covered either due to RCIC operation or automatic or manual depressurization permitting low pressure inventory makeup.

Criterion 3 is related only to accidents and not AOOs.

10 CFR 50 Appendix A Criterion 1:

General Design Criterion 1 (GDC 1) of 10 CFR 50 Appendix A – "Quality standards and records" includes the following general requirements

> Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

NUREG-0138 addressed 15 technical issues including the treatment of non-safety grade equipment in evaluations of postulated steam line break accidents (Issue No. 1). The staff's evaluation concluded that GDC 1 expressly permits flexibility in the acceptance level for safety related equipment and that less stringent quality standards for certain "non-safety grade" equipment was appropriate. Additionally, in a letter dated August 22, 1977 (Reference 5), regarding the use of non-safety grade equipment to mitigate the consequences of a tube rupture accident, the NRC staff referred to Issue No. 1 of NUREG-0138 in reaching the following conclusion:

> As indicated in Issue No. 1, NRR weighs the consequences of accidents and safety requirements in order to assure a balanced level of safety over the entire spectrum of postulated design basis accidents. Where potential consequences are at the low end of the spectrum, NRR imposes less stringent requirements on quality and design of systems and components. In some instances, NRR accepts reliance on "non safety grade" equipment for mitigation of consequences of accidents.

These previous NRC positions indicate that the regulations do not specifically exclude non safety equipment from being credited in AOO analyses. NRC guidance provided in the Standard Review Plan also recognizes the possibility of crediting non-safety functions as discussed below.

Standard Review Plan, NUREG 0800:

The Standard Review Plan (SRP) NUREG 0800, Rev. 1 specifically includes a discussion of the loss of normal feedwater flow event in section 15.2.7. The SRP provides instructions on the review of such events which include the following items that recognize the possible use of non-safety systems in the LOF analysis:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

Therefore, the staff's position that the regulations allow only safety-grade systems to be credited for mitigating an AOO is not consistent with the above guidance.

GGNS Licensing Basis

The plant response to design basis AOOs including the LOF are discussed in the GGNS licensing basis.

Evaluations of AOOs take credit for the active functioning of normally operating plant systems, including some "non-safety-related" systems. Although not controlled by Tech Specs, functionality of non-safety-related systems is required for plant operation and their failure would be detected in the control room. A specific example is the main turbine Initial Pressure Control (IPC) system and Electro-Hydraulic Controller (EHC). The IPC system provides for reactor pressure regulation during all modes of operation by positioning the four High Pressure Turbine Control Valves (TCVs). Normal functioning of the IPC system is credited in the safety analyses. Three specific examples of AOO's described in the GGNS UFSAR that credit the functioning of the IPC system and turbine control valves are discussed below.

Loss of Feedwater Heating (LOFH) (UFSAR 15.1.1): The LOFH is initiated by a postulated loss of a high pressure feedwater heater resulting in a gradual reduction in FW temperature of 100°F. The analysis assumes that the time for the change in FW temperature is greater than 80 seconds based on the stroke times of the FW heater bypass valves and thermal inertia of the FW system. The reduction in FW temperature results in a slow increase in core inlet subcooling, a decrease in core average void fraction, and increase in reactor power due to the negative void reactivity coefficient. Because the LOFH event is a slow event, the safety analysis assumes quasi-steady-state conditions and evaluates this event using the fuel vendor's steady-state BWR simulator code (PANACEA) as approved by GESTAR II (Ref. 2). The intrinsic assumption made when using steady-state methodology is that the final steady-state minimum critical power ratio (MCPR) is less than the MCPR occurring during the transient phase of the event. *This is only possible if normal functioning of the IPC system is assumed.* The IPC responds to the increased core power and steam flow and opens the TCVs fast enough to avoid core pressurization. Note also that normal functioning of the Feedwater Control System (FWCS) is also implicitly assumed in this analysis such that the reactor responds smoothly to the increases in reactor power.

Single Loop Flow Run-up (UFSAR 15.4.5): The single loop flow run-up transient considers the inadvertent opening of one FCV at its minimum stroking rate. The resulting slow core flow excursion results in an increase in core power and steam flow. The flow-dependent MCPR operating limits (MCPRf) and LHGR multipliers (LHGRFACf) are set by the flow run-up transient. Similar to the LOFH event, the slow flow run-up is analyzed assuming quasi-steady-state conditions using the BWR simulator code. The IPC system is thus credited to respond to the change in reactor conditions by opening the TCV's fast enough to avoid core pressurization during the flow increase phase of the event. Normal functioning of the

Feedwater Control System (FWCS) is also implicitly assumed in this analysis such that the reactor responds smoothly to the increases in reactor power.

Feedwater Controller Failure-Maximum Demand (FWCF, UFSAR 15.1.2): The FWCF is initiated by a postulated failure in the FWCS to maximum demand resulting in a rapid increase in FW flow. Although this is not a slow or quasi-steady-state event, normal functioning of the IPC system is assumed. The FWCF consists of an overcooling stage during the feedwater flow run-up followed by a fast pressurization/turbine trip stage when vessel water level reaches the high level trip. During the overcooling stage, the cooler water introduced into the core results in a small power increase. Steam flow also increases, but not at the same rate as power since the lower core inlet temperature reduces the steam generation. The IPC system is assumed to respond to the increased steam flow by opening the TCVs slightly from the initial position. Without this assumption, the FWCF event would become more severe due to pressurization before the high water level trip.

In summary, the GGNS safety analyses include moderate frequency events (AOO's) that require and assume normal functioning of the non-safety-related main turbine initial pressure control system and the feedwater control system to ensure that the reactor core safety limits are not exceeded. This is acceptable because the functionality of these systems is required for normal plant operation. The current GGNS Licensing Basis is not clear concerning the credit taken for the CST in the LOF AOO analysis since the analysis stops at HPCS initiation, although it is clear that the non safety feedwater system remains intact and delivers its remaining water flow to the vessel during the event. Therefore, in the letter dated February 14, 2008 (Reference 3), GGNS proposed crediting the integrity of the non-safety grade Condensate Storage Tank (CST) and suction piping during a Loss of Feedwater event (LOF, UFSAR 15.2.7). Specifically, in discussing the LOF event response, it was stated that "...CST inventory is sufficient to accommodate the inventory requirements for the SBO event for the coping period of 4 hours" and further, that "...assuming an initial [CST] 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..." (Reference 3, Attachment 1, page 3, paragraph 4). That is, the CST is assumed to be available to provide the required water inventory for HPCS to restore and maintain water level above the TS 2.1.1.3 safety limit (TAF) such that the CST suction swap would not be required. The assumption of normal functioning of the non-safety-grade CST in this scenario is consistent with the crediting of the IPC system in the current GGNS safety analyses described above.

NRC Comment 1:

In section 2.0 of the denial evaluation, it is stated that "There are no requirements in the TSs on the minimum volume of water in the CST for Modes 1, 2, and 3;".

Response:

This statement is correct, however a TS limit is not necessary because the level needed for HPCS and RCIC is protected by the design of the tank. Standpipes inside the CST ensure that the non-safety systems cannot draw the CST level below 18.9 feet.

NRC Comment 2:

In section 4.2 of the denial evaluation, it stated that "UFSAR Section 6.3.1.1.1.d states that the HPCS system is provided for maintaining the water above the top of the core. "

Response:

This statement is incomplete and not applicable to the subject request. UFSAR Section 6.3.1.1.1.c states "One high-pressure cooling system is provided which is capable of maintaining water level above the top of the core and preventing ADS actuation for breaks of lines less than 1 inch nominal diameter." The statement is associated with the Small Break LOCA design basis accident. The concept of safety-limit related and avoiding the safety limits of TS 2.1 do not apply to accidents as these events are evaluated against the radiological consequences of 10 CFR 50.67.

4.0 Conclusion:

Entergy maintains that the setpoint is not SL-related. The non-safety CST and associated piping provide water to the HPCS system to maintain the reactor vessel water level safety limit during the loss of feedwater transient. Failure of these passive components need not be assumed to fail during the loss of feedwater event since its failure would be beyond an event of moderate frequency. Even in the unlikely event of piping or tank failure, the unit can be maintained with adequate inventory by automatic or manual depressurization with low pressure ECCS system inventory makeup.

Entergy also contends that the requested change should not be denied based upon the NRC staff positions presented in RIS 2006-17. Entergy believes that these positions may constitute new requirements that need further regulatory review and processing. The requested change is needed to maintain the current GGNS licensing basis conservative with respect to the design of the plant. Denial of the request would leave the GGNS TS requirement to be non-conservative. Entergy believes that safety and the stability of the regulatory process would best be served by promptly approving the amendment request while allowing industry activities to reach a generic solution of the issues discussed in RIS 2006-17.

5.0 References:

1. Regulatory Guide (RG) 1.70, Rev. 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," November, 1978.
2. NEDE-24011-P-A, *General Electric Standard Application for Reactor Fuel (GESTAR-II).*
3. GNRO-2008/00006, W. R. Brian (Entergy) to USNRC, "Supplement 3 to Amendment Request, Condensate Storage Tank Setpoint Change" dated February 14, 2008.
4. Michael T. Markley (USNRC) to Entergy, "Grand Gulf Nuclear Station Unit 1 – Staff Evaluation and Basis for Denial of Proposed Technical Specification Change Related to the Condensate Storage Tank Level-Low Setpoint Changes" dated November 5, 2008.
5. NRC memorandum from K. V. Seyfrit, Assistant Director, Technical Programs, ROI, IE to R. F. Warnick, Chief, Reactor Projects Section 2, RIII – Action Item F30310H2, Westinghouse Tube Rupture Accident.