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GNRO-2003/00032

May 12, 2003

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

SUBJECT: Grand Gulf Nuclear Station, Unit 1  
Docket No. 50-416  
License Amendment Request  
Shutdown Cooling System Isolation Instrumentation

REFERENCE: Letter from U.S. NRC to Mr. William T. Cottle, Grand Gulf Nuclear  
Station, Unit 1, - Issuance of Amendment RE: Cold Shutdown and  
Refueling Conditions (TAC No. 76758), dated September 24, 1990.

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the following amendment for Grand Gulf Nuclear Station, Unit 1 (GGNS). Entergy requests modification of GGNS Technical Specification (TS) 3.3.6.1 "Primary Containment and Drywell Isolation Instrumentation" to add a provision to the APPLICABILITY function that will eliminate the requirement that the Residual Heat Removal (RHR) System Isolation, Reactor Vessel Water Level - Low, Level 3, be OPERABLE under certain conditions during refueling outages. Specifically, the proposed change removes the requirement for this isolation function, specified in Table 3.3.6.1-1, when the upper containment reactor cavity is at the High Water Level (HWL) condition specified in TS 3.5.2, Emergency Core Cooling Systems (ECCS) Shutdown. The proposed changes would allow various surveillances and other outage activities to be completed efficiently during refueling by eliminating the risk associated with an unintended or spurious isolation that would result in a loss of the shutdown cooling function.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

The proposed changes include no new commitments. Proposed changes to the Technical Specification Bases are included as Attachment 3 for information only. Entergy requests approval of the proposed amendment by January 30, 2004. Once approved, the amendment shall be implemented within 60 days. Although this request is neither exigent nor emergency, your prompt review is requested.

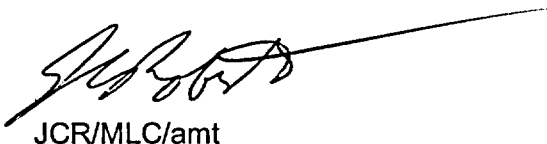
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May 12, 2003  
GNRO-2003/00032  
Page 2 of 2

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 12, 2003.

Sincerely,



JCR/MLC/amt

Attachments:

1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. Changes to Technical Specification Bases Page (For Information)

cc: U.S. Nuclear Regulatory Commission  
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Mr. N. S. Reynolds  
Mr. H. L. Thomas

**Attachment 1**

**GNRO-2003/00032**

**Analysis of Proposed Technical Specification Change**

## 1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-29 for Grand Gulf Nuclear Station, Unit 1 (GGNS).

Entergy requests changes to Section 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation" of the GGNS Technical Specifications (TS), Appendix A of the Operating License. Specifically, the proposed change removes the Level 3 (Reactor Vessel Low Water Level) Residual Heat Removal (RHR) system isolation requirement when the upper containment reactor cavity is at the High Water Level (HWL) condition specified in TS 3.5.2, Emergency Core Cooling Systems (ECCS) Shutdown.

The purpose of the proposed change is to allow certain outage-related activities to be performed efficiently without an undue burden on operations personnel resources and to minimize the risk of spurious or unintended shutdown cooling isolations. The next GGNS refueling outage is scheduled for the first quarter of 2004. Entergy desires that this amendment be issued by January 30, 2004 to support work planning prior to the outage.

## 2.0 PROPOSED CHANGE

Primary containment and drywell instrumentation operability requirements are governed by TS LCO 3.3.6.1, Primary Containment and Drywell Isolation Instrumentation. Applicability requirements for this instrumentation are specified in Table 3.3.6.1-1. In MODE 5, the RHR System Isolation on Low Reactor Vessel Water Level (Table 3.3.6.1-1, Item 5.b) is required at all times. The number of trip systems is modified by footnote "d" which requires only one trip system in MODES 4 and 5 with RHR Shutdown Cooling (SDC) System integrity maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system. The proposed change adds an additional footnote for MODE 5 that states that the function is not required with the upper containment reactor cavity and transfer canal gates removed and water level  $\geq 22$  ft 8 inches over the top of the reactor pressure vessel flange. The proposed footnote is only applicable in MODE 5. Note that the cavity gate, transfer gate and water level requirements match those for ECCS operability in this condition specified by TS 3.5.2.

Thus, Table 3.3.6.1-1 will be revised as follows:

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS
5. RHR System Isolation  b. Reactor Vessel Water Level – Low, Level 3	5 <sup>(g)</sup>
(g) Not applicable with the upper containment reactor cavity and transfer canal gates removed and water level $\geq 22$ ft 8 inches above the reactor pressure vessel flange.	

In summary, Entergy is proposing to revise the operability requirements for the RHR system isolation instrumentation on low reactor vessel water level when the upper containment reactor cavity is at the high water level condition specified in TS 3.5.2, ECCS Shutdown.

Associated changes to the TS Bases are provided in Attachment 3. The proposed TS Bases changes are for information only and will be controlled by TS 5.5.11, "Technical Specifications Bases Control Program."

### 3.0 BACKGROUND

During refueling outages, a number of operational and instrumentation surveillance tests are conducted that have the potential to actuate the isolation logic causing one or both of the Shutdown Cooling (SDC) isolation valves (E12F008 and E12F009) to automatically close. The closure of either isolation valve interrupts shutdown cooling system operation. Although this occurrence has been infrequent and is recoverable, a loss of SDC causes a significant challenge to operators to respond to the event.

The complexity of the isolation logic itself has contributed to spurious isolations in the past. The logic uses four power sources in four different panels. Each isolation valve is potentially actuated by two divisions of logic which include contacts and logic for isolation functions that are not required to be operable during MODE 5 (e.g., High Reactor Vessel Pressure). As a result of this arrangement, operations personnel must expend considerable time and effort to prevent spurious actuations (i.e., lifted leads, etc.) and the refueling outage schedule is adapted (alternate flow path) to accommodate the maintenance periods where the logic is potentially affected. Additionally, these efforts pose some risk of causing an isolation.

To optimize outage scheduling, free personnel resources for other activities, and assure shutdown cooling system availability, GGNS desires to inhibit the automatic isolation function by opening the breakers to the E12F008 and E12F009 isolation valves. By implementing the proposed changes to the Technical Specifications, testing and other outage-related work can be performed without the need to lift leads in the isolation logic increasing scheduling flexibility and saving outage resources.

### 4.0 TECHNICAL ANALYSIS

#### 4.1 Description of Current Requirements

TS Section 3.3.6.1 LIMITING CONDITION FOR OPERATION (LCO) requires low reactor water level instrumentation and isolation logic associated with the RHR system isolation to be operable at all times in MODE 5. The instrumentation operability invokes an operability requirement for the associated containment isolation valves (1E12F008 and 1E12F009) under specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs).

#### 4.2 Bases for Current Requirements

As discussed in the bases for TS Section 3.3.6.1, the low reactor vessel water level isolation is not directly assumed in any transient or accident analyses during MODE 5. This function simply supports actions to ensure that the RPV water level does not drop below the top of the active fuel during a vessel draindown event through 1E12F008 and 1E12F009 caused by a leak (i.e.,

pipe break or inadvertent valve opening) in the RHR SDC System. During MODE 5, a low reactor water level signal could indicate that inventory is being lost through a breach in the shutdown cooling or connected piping or that a drain path has been inadvertently created through an improper valve manipulation. The primary function of the automatic isolation of the RHR SDC isolation instrumentation for vessel Level 3 (TS Table 3.3.6.1-1 function 5.b) is to terminate inventory losses through the shutdown cooling flow path by closing 1E12F008 and 1E12F009. This action will prevent the reactor core from becoming uncovered and potentially overheated as reactor inventory diminishes. The isolation function is initiated at Level 3, the highest low water level setting. This function works in conjunction with the ECCS to mitigate reactor vessel draindown events through all drainage paths. A diagram showing the various level settings is provided in TS Bases Figure B 3.3.1.1- 1.

During MODE 5, the water inventory available to be lost through any given drain path can vary significantly. With the reactor cavity drained and water level below the reactor vessel flange elevation, a draindown event could lead to a low water level (Level 3) condition in a relatively short time. Since this configuration is more limiting than with the reactor cavity flooded, this was used to evaluate draindown events as part of the licensing actions for the Alternate Decay Heat Removal System (ADHRS). As a result of these evaluations and as part of the ADHRS changes to the GGNS TS, Entergy (formerly SERI) requested and received the TS changes that added the current requirement for an automatic isolation of the SDC suction line (Reference).

In the analyses supporting these changes, various flow paths were evaluated assuming a draindown event is initiated by a single operator error or equipment malfunction. Note that these analyses considered the initial operator awareness of a draindown event at the low level (Level 3) alarm. Several drain paths were eliminated since they were essentially self limiting, that is, the event would terminate without any action based only on the associated piping configuration (e.g., drain paths through the feedwater lines). Several potential drain and pump-down paths with relatively low flow rates were deemed acceptable with no credit for automatic isolation based on the criterion that operators had greater than 20 minutes to isolate the drain path and realign ECCS and inject into the reactor vessel after detection by the control room of the inventory loss at Level 3. The remaining drain paths would satisfy the criterion only by crediting the Level 3 automatic SDC isolation. Hence, the requirement for automatic isolation of the SDC flow path was requested.

The flow path relying on the Level 3 isolation identified with the highest flow rate is a pump-down path from the reactor vessel to the suppression pool via the minimum flow line. This flow path takes suction from the recirculation loop through 1E12F008 and 1E12F009 and discharges through the RHR minimum flow path to the suppression pool. The flow path is created if the RHR minimum flow valve (1E12F064 A/B) fails to close during startup of the shutdown cooling loop. The flow rate through this flow path was determined to be 1435 gallons per minute with the initial water level at the reactor vessel flange.

The requirement for operability of the RHR SDC isolation function on low water level was added to the GGNS TS in Amendment 70 in conjunction with licensing actions for the ADHRS. The ADHRS was designed and built to supplement the RHR SDC mode during Operational Condition (OC) 4, "Cold Shutdown," and OC 5, "Refueling." The requirements for this function were subsequently added generically to the improved Standard Technical Specifications (STS), NUREG-1434. This request deviates from the improved STS requirements for the isolation function during high water level based on a plant specific evaluation.

#### 4.3 Safety Analysis of Proposed Changes

The proposed change eliminates the requirement for the automatic isolation of 1E12F008 and 1E12F009 at low water level 3 provided the upper containment reactor cavity and transfer canal gates are removed and water level is  $\geq 22$  ft 8 inches over the top of the reactor pressure vessel flange (HWL). The proposed changes are only applicable in MODE 5. Operability requirements for ECCS are unrelated to the proposed changes since protective actions are expected to occur well before ECCS initiation.

As mentioned above, the water inventory available to be lost through any given drain or pump-down path can vary significantly during MODE 5. With the reactor cavity flooded, a worst case pump-down event described above with no mitigating actions would take considerable time to reach the Level 3 isolation setpoint. Building on the draindown analysis performed for the previous TS changes discussed above, the same flow path with the upper containment reactor cavity at the HWL results in a flow rate through the RHR minimum flow line of approximately 1450 gallons per minute (gpm). Accounting for the additional water inventory available with these pools flooded and the gates removed, an inventory loss of 1450 gpm would not reduce the pool level to the reactor flange for approximately 4½ hours. If the inventory associated with an equivalent loss of level took credit for adjoining pools in the auxiliary building, the time would be significantly longer. Given this extended period for operator detection and response to a draindown event, sufficient time is available before reaching the automatic isolation setpoints previously associated with Reactor Low Water Level 3 and for operations personnel to take action to manually close either the E12F008 or 1E12F009 valves or to terminate the inventory loss by other means (e.g., closing the RHR minimum flow valve) prior to uncovering fuel.

Several methods are readily available to identify an event where significant inventory is being lost. These include the following:

- The Fuel Pool Drain tank level is monitored and alarms on low level in the drain tank. This would be one of the primary means to identify a loss of inventory, providing an early alarm.
- With the large contingent of people on the refuel floor (fuel movers, reactor engineers, SROs) during refueling outages, it is reasonable to expect that the falling water level in the pools would be noticed well before it reaches the vessel flange and that the control room would be notified.
- Although periodically defeated for maintenance, surveillances, etc., undervessel sumps are equipped with an alarm function and a large influx of water would cause the alarms to annunciate.
- Assuming irradiated fuel is stored in the upper pool, a loss of level would cause Area Radiation Monitor alarms. (15 mR)
- The HPCS Level 8 alarm is typically sealed in due to HWL in the reactor cavity. This alarm will clear and slow flash if level drops leading a control room operator to tie this to a lowering of water level.

- The reactor vessel high water level alarm clears at 56 inches indicating inventory loss notifying operators.

The diverse methods available described above to recognize that a draindown event has occurred and the relatively long period of time available to respond to such an event is consistent with the GGNS licensing basis to terminate a draindown event.

As discussed in the bases for TS 3.5.2, ECCS Shutdown, draindown events in MODE 5 with the reactor cavity flooded are not a concern since the condition "provides sufficient coolant inventory to allow operator action to terminate the inventory loss prior to fuel uncover in case of an inadvertent draindown." As shown above, this capability continues to be the case without the SDC suction flow path Level 3 isolation function. As a result, inoperability of the shutdown cooling suction flow path automatic isolation, in itself, is not a condition where a draindown event could create the potential for the release of fission products. Since radiological releases are not postulated to occur due to the large water inventory and manual isolation capability, additional systems used to mitigate radiological releases such as those that apply during operations with an increased potential for draining the reactor vessel, need not be invoked during this condition.

## 5.0 REGULATORY ANALYSIS

### 5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. The application provides sufficient information to demonstrate that the request does not alter compliance with any applicable regulatory requirement or criteria. The containment isolation function (GDC 55 and 56) that the valves provide is only applicable in MODES 1, 2, and 3 consistent with the requirement for containment integrity. Therefore, there is no change affecting the containment isolation function.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any GDC differently than described in the Final Safety Analysis Report (FSAR).

### 5.2 No Significant Hazards Consideration

Pursuant to 10 CFR 50.90, Entergy Operations Inc., (Entergy) hereby requests amendment of Facility Operating License for Grand Gulf Nuclear Power Station (GGNS). Specifically, Entergy requests to add a footnote to Table 3.3.6.1-1 item 5.b for MODE 5 that states that the function is not required with the upper containment reactor cavity and transfer canal gates removed and water level  $\geq 22$  ft 8 inches over the top of the reactor pressure vessel flange. The proposed footnote is only applicable in MODE 5.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:



1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change revises the applicability requirement for the Residual Heat Removal (RHR) System Isolation function of the Primary Containment and Drywell Isolation Instrumentation during MODE 5. The change removes the requirement that the instrumentation be operable during certain conditions during refueling outages. The function is intended to mitigate reactor vessel draindown events. Although draindown events during refueling operations are not specifically evaluated in the Updated Final Safety Analysis Report (UFSAR), these events were evaluated in support of licensing actions for the Alternate Decay Heat Removal System (ADHRS). The probability that a draindown event will be initiated is unrelated to operability requirement for this instrumentation or the associated isolation valves. The evaluation supporting this change determined that mitigating actions can be taken to terminate all postulated draindown events prior to fuel uncover. As a result, the probability of draindown events causing fuel uncover and the potential for radiological releases has not significantly increased. The operation or failure of the shutdown cooling suction isolation does not contribute to the occurrence of an accident. No active or passive failure mechanisms that could lead to an accident are affected by the proposed change.

The consequences of a vessel drainage event are not significantly increased by the proposed change. Entergy has evaluated various draindown and pumpdown events through the shutdown cooling flow path and determined that adequate time is available for operations personnel to identify and take action to mitigate such events such that adequate core cooling is maintained and a radiological release does not occur.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Entergy has evaluated various draindown events through the shutdown cooling flow path and determined that adequate time is available for operations personnel to identify and take action to mitigate any events such that adequate core cooling is maintained. With the containment refueling cavity flooded, sufficient inventory is available to allow operator action to terminate the inventory loss prior to reaching a low water level in the reactor. Installed equipment is not operated in a new or different manner; no new or different system interactions are created, and no new processes are introduced. No new failures have been created by the proposed changes.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes do not introduce any new setpoints at which protective or mitigative actions are initiated. No current setpoints are altered by this change. The design and functioning of the containment and drywell isolation function is also unchanged. The change simply modifies the applicability of the Technical Specifications (TS) by removing the requirement that the RHR system isolation on low reactor vessel level be operable with the upper containment cavity flooded in MODE 5. During MODE 5, the RHR system isolation mitigates postulated draindown events through the RHR system. Entergy has evaluated various draindown events through this flow path and determined that adequate time is available for operations personnel to identify and take action to mitigate such events such that adequate core cooling is maintained.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

### 5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

**Attachment 2**

**GNRO-2003/00032**

**Proposed Technical Specification Changes (mark-up)**

Primary Containment and Drywell Isolation Instrumentation  
3.3.6.1

Table 3.3.6.1-1 (page 5 of 5)  
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
<b>5. RHR System Isolation</b>					
a. RHR Equipment Room Ambient Temperature - High	1,2,3	1 per room	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 171°F
b. Reactor Vessel Water Level - Low, Level 3	1,2,3 <sup>(e)</sup>	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 12.8 inches
	3 <sup>(f)</sup> , 4, 5 <sup>(g)</sup>	2 <sup>(d)</sup>	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 10.8 inches
c. Reactor Steam Dome Pressure - High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 150 psig
d. Drywell Pressure - High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 1.43 psig
e. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.7	NA

(d) Only one trip system required in MODES 6 and 5 with RHR Shutdown Cooling System Integrity maintained.

(e) With reactor steam dome pressure greater than or equal to the RHR cut-in permissive pressure.

(f) With reactor steam dome pressure less than the RHR cut-in permissive pressure.

(g) Not applicable with the upper containment reactor cavity and transfer canal gates removed and water level ≥ 22 ft 8 inches above the reactor pressure vessel flange.

**Attachment 3**

**GNRO-2003/00032**

**Changes to Technical Specification Bases Page (For Information)**

**Insert at the end of the first paragraph on page B 3.3-160**

When neither trip system is required OPERABLE in MODE 5, the applicable safety analysis assumes that the RHR shutdown cooling valves are easily recoverable such that a draindown event through the shutdown cooling flow path can be terminated by manually closing one or more of the Shutdown Cooling System isolation valves. This condition, in itself, is not considered an OPDRV.

Primary Containment and Drywell Isolation Instrumentation  
B 3.3.6.1

BASES

APPLICABLE  
SAFETY ANALYSES,  
LCO, and  
APPLICABILITY

5.b. Reactor Vessel Water Level—Low, Level 3 (continued)

Water Level—Low, Level 3 Function are required to be OPERABLE in MODES 4 and 5 (both channels must input into the same trip system) provided the RHR Shutdown Cooling System integrity is maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system. When only one trip system is OPERABLE in MODE 4 or 5, the trip system should be considered inoperable if the associated RHR Shutdown Cooling System suction from the reactor vessel isolation valve (i.e., the 1E12-F008 or 1E12-F009) is not associated with an OPERABLE diesel generator. ←

The Reactor Vessel Water Level—Low, Level 3 Allowable Value was chosen to be the same as the RPS Reactor Vessel Water Level—Low, Level 3 Allowable Value (LCO 3.3.1.1) since the capability to cool the fuel may be threatened.

The Reactor Vessel Water Level—Low, Level 3 Function is required to be OPERABLE in MODE 3 with reactor pressure less than the RHR permissive pressure, MODE 4, and MODE 5 to prevent this potential flow path from lowering reactor vessel level to the top of the fuel. This instrumentation is required to be OPERABLE in MODES 1 and 2 and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut-in permissive pressure to support actions to ensure that offsite dose limits of 10CFR100 are not exceeded.

This function isolates the Group 3 valves.

5.c. Reactor Steam Dome Pressure—High

The Shutdown Cooling System Reactor Steam Dome Pressure—High Function is provided to isolate the shutdown cooling portion of the RHR System. This interlock is provided only for equipment protection to prevent an intersystem LOCA scenario and credit for the interlock is not assumed in the accident or transient analysis in the UFSAR.

(continued)