Entergy

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

George A. Williams
Vice President - Operations
Grand Gulf Nuclear Station

GNRO-2003/00072

December 5, 2003

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

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

REFERENCES:

1. Letter from USNRC 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.

2. Letter GNRO-2003/00032 from Mr. Jerry C. Roberts to USNRC, Grand Gulf Nuclear Station, Unit 1 - License Amendment Request, Shutdown Cooling System Isolation Instrumentation, dated May 12, 2003.

Dear Sir or Madam:

By letter dated May 12, 2003 (Reference 2), Entergy Operations, Inc. (Entergy) proposed changes to Grand Gulf Nuclear Station, Unit 1 (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 would remove 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.

Entergy and members of your staff held several calls to discuss the proposed changes. After discussion and review, Entergy agreed to reword the isolation provision and add a surveillance requirement that would enhance the ability of operations personnel to detect potential inventory losses. Revisions of the TS analysis in the original submittal

A001

(Reference 2) are indicated with change bars in the margin of Attachment 1 and the associated marked up TS pages are provided in Attachment 2. Changes to the TS Bases associated with these changes are provided in Attachment 3 for your information and will be implemented in accordance with TS 5.5.1-1, "Technical Specification Bases Control Program."

The proposed revision to the original submittal 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 these changes involve no significant hazards consideration. The bases for these determinations are included in the attached submittal.

No new commitments are contained in this letter. 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.

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 December 5, 2003.

Sincerely,

George A. Williams

GAW/MLC/amt

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

cc: (See Next Page)

cc: U.S. Nuclear Regulatory Commission
ATTN: Mr. Bruce S. Mallett (w/2)
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

U.S. Nuclear Regulatory Commission
ATTN: Mr. Bhalchandra Vaidya, NRR/DLPM (w/2)
ATTN: ADDRESSEE ONLY
ATTN: U.S. Postal Delivery Address Only
Mail Stop OWFN/7D-1
Washington, D.C. 20555-0001

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

Mr. T. L. Hoeg, GGNS Senior Resident
Mr. D. E. Levanway (Wise Carter)
Mr. L. J. Smith (Wise Carter)
Mr. N. S. Reynolds
Mr. H. L. Thomas

Attachment 1

GNRO-2003/00072

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 a new surveillance requirement (SR 3.3.6.1.9) to verify the water level in the upper containment pool is ≥ 22 ft 8 inches above the reactor pressure vessel flange every four hours and 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 when the upper containment reactor cavity and transfer canal gates are removed and SR 3.3.6.1.9 is met. The proposed SR and 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	SURVEILLANCE REQUIREMENTS
5. RHR System Isolation		
b. Reactor Vessel Water Level – Low, Level 3	5[g]	SR 3.3.6.1.9

A new surveillance is proposed as follows:

Surveillance		Frequency
	------------------NOTE------------------ Only required to be performed when Function 5.b is not OPERABLE as allowed by Note (g) of Table 3.3.6.1-1. --	
SR 3.3.6.1.9	**Verify the water level in the Upper Containment Pool is ≥ 22 feet, 8 inches above the reactor pressure vessel flange.**	**4 hours**

In summary, Entergy is proposing to revise the operability requirements for the Residual Heat Removal 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, "Emergency Core Cooling Systems Shutdown." Entergy is also proposing to add a surveillance requirement to enhance the ability of operations personnel to detect a loss of inventory.

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 TS Section 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 1).

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). Other flow paths were eliminated since they required either multiple operator errors to establish or the error was determined to be not credible given the plant configuration, administrative barriers and normal operating practices. 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 ≥ 22 ft 8 inches above the top of the reactor pressure vessel flange (HWL). In addition, Entergy proposes a new surveillance requirement to verify the water level in the upper containment pool every four hours. 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 reenergize and 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. To enhance the ability of operations personnel to detect inventory loss associated with a draindown event an upper containment pool surveillance requirement is proposed.

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 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. An additional evaluation supporting this change established that the RHR system automatic isolation was not needed to mitigate a draindown event given the possible drain paths and the time available for operators to terminate the 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 uncovery 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 <u>Applicable Regulatory Requirements/Criteria</u>

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 new surveillance requirement (SR 3.3.6.1.9) to verify the water level in the Upper Containment Pool every four hours and 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 when the upper containment reactor cavity and transfer canal gates are removed and water level is ≥ 22 ft 8 inches above the top of the reactor pressure vessel flange (SR 3.3.6.1.9 is met). The proposed SR and footnote are only applicable in MODE 5.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment 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 and adds a surveillance requirement that is invoked when specific conditions exist. The proposed surveillance requirement only enhances the ability of operating personnel to detect inventory loss associated with a draindown event. The change removes the requirement that the instrumentation be operable during certain conditions (high water level) during refueling outages. The isolation function is intended to mitigate reactor vessel draindown events by isolating the residual heat removal flow path at low reactor water level. 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. An additional evaluation supporting this change established that the RHR system automatic isolation was not needed to mitigate a draindown event given the possible drain paths and the time available for operators to terminate the draindown event. The probability that a draindown event will be initiated is unrelated to operability requirement for this instrumentation, the associated isolation valves or the proposed surveillance. The evaluation determined that mitigating actions can be taken to identify and terminate all postulated draindown events prior to fuel uncovery. As a result, the probability of draindown events causing fuel uncovery 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. The proposed surveillance requirement only enhances the ability of operating personnel to detect inventory loss associated with a draindown event. 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 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. The proposed surveillance requirement only enhances the ability of operating personnel to detect inventory loss associated with a draindown event and does not impact a margin of safety. 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-2003/00072

Proposed Technical Specification Changes (mark-up)

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
5. RHR System Isolation					
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(e)	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.7	≥ 10.8 inches
	3(f), 4, 5 (g)	2(d)	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.7 SR 3.3.6.1.9	≥ 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 3.3.6.1.4 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 3.3.6.1.4 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 4 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 when the upper containment reactor cavity and transfer canal gates are removed, and SR 3.3.6.1.9. Is met.

SURVEILLANCE REQUIREMENTS

NOTES

1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.1.3	Calibrate the trip unit.	92 days
SR 3.3.6.1.4	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.6.1.5	Perform CHANNEL CALIBRATION.	12 months
SR 3.3.6.1.6	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.1.7	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.6.1.8	NOTE: Channel sensors may be excluded. Verify the ISOLATION SYSTEM RESPONSE TIME for the Main Steam Isolation Valves is within limits.	18 months on a STAGGERED TEST BASIS

INSERT A

INSERT "A"

	Surveillance	Frequency
	------------------NOTE------------------ Only required to be performed when Function 5.b is not OPERABLE as allowed by Note (g) of Table 3.3.6.1-1. --	
SR 3.3.6.1.9	**Verify the water level in the Upper Containment Pool is ≥ 22 feet, 8 inches above the reactor pressure vessel flange.**	**4 hours**

Attachment 3

GNRO-2003/00072

Changes to Technical Specification Bases Page (For Information)

<u>Insert at the end of the first paragraph on page B3.3-160</u>

When neither trip system is required to be OPERABLE in MODE 5, the applicable safety analysis assumes that the RHR shutdown cooling isolation valves are easily recoverable (such as by maintaining at least one valve capable of being remotely closed after reenergizing the valve) such that a drain down event through the shutdown cooling flow path can be terminated prior to reaching Level 3 by closing one or more of the Shutdown Cooling System isolation valves. Based on the analysis of the time available to mitigate all postulated drain down events, this condition, in itself, is not considered an OPDRV.

<u>Insert following SR 3.3.6.1.8 on page B3.3-170a</u>

<u>SR 3.3.6.1.9</u>

Analysis has shown that with the upper containment pool cavity flooded and the gates removed, adequate time exists to allow operator action necessary to terminate the inventory loss prior to reaching reactor level 3. This analysis takes credit for the pool level being greater than or equal to 22 feet 8 inches above the reactor vessel flange. Verifying the upper containment pool level is greater than or equal to 22 feet 8 inches on a four hour frequency provides assurance that the operators have enough time to detect and terminate a drain down event.

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.

Insert here

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)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.8 (continued)

instrument response times and assure operation of the analyzed instrument loops within acceptable limits. Reference 7 also identifies that there are no known channel sensor failure modes identified that can be detected by response time testing that cannot also be detected by other Technical Specification required surveillances. Therefore, when the requirements, including sensor types, of Reference 7 are complied with, adequate assurance of the response time of the sensors is provided. This assurance of the response time of the sensors when combined with the response time testing of the remainder of the channel ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The calibration shall be performed such that fast ramp or step change to system components during calibrations is performed to verify that the response of the transmitter to the input change is prompt. Technicians shall monitor for response time degradation during the performance of calibrations. Technicians shall be appropriately trained to ensure they are aware of the consequences of instrument response time degradation. These items are commitments made per Reference 8. If the alternate testing requirements of Reference 7 are not complied with then the entire channel will be response time tested including the sensors.

ISOLATION SYSTEM RESPONSE TIME tests for this instrumentation are conducted on an 18 month STAGGERED TEST BASIS. This test Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience that shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent.

Insert here

REFERENCES

1. UFSAR, Chapter 6.
2. UFSAR, Chapter 15.

(continued)