

Function on Reactor Vessel Water Level-Low, Level 3, be OPERABLE during certain conditions in MODE 5. Specifically, the proposed change would remove the requirement when the upper containment reactor cavity is at the High Water Level condition specified in TS 3.5.2, "Emergency Core Cooling Systems (ECCS) Shutdown."

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented 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 [Entergy Operations, Inc.] 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.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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*Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Entergy Mississippi, Inc., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi*

*Date of amendment request:* May 12, 2003.

*Description of amendment request:* The proposed amendment would change administrative Technical Specification (TS) 5.5.12 regarding containment integrated leakage rate

testing (ILRT) and TS 3.6.5.1.1 regarding drywell bypass leak rate testing (DWB). The change would allow for a one-time extension of the interval (15 years) for performance of the next ILRT and DWBT.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

*Response:* No.

The proposed amendment to TS 5.5.12 adds a one-time extension to the current interval for Type A testing (*i.e.*, the ILRT) and the DWBT. The current interval of ten years, based on past performance, would be extended on a one-time basis to 15-years from the date of the last test. The proposed extension to the Type A test cannot increase the probability of an accident since there are no design or operating changes involved and the test is not an accident initiator. The proposed extension of the test interval does not involve a significant increase in the consequences since research documented in NUREG-1493, "Performance Based Containment Leak Rate Test Program," has found that, generically, fewer than 3% of the potential containment leak paths are not identified by Type B and C testing. A risk evaluation of the interval extension for GGNS [Grand Gulf Nuclear Station, Unit 1] is consistent with these results. In addition, the testing and containment inspections also provide a high degree of assurance that the containment will not degrade in a manner detectable only by a Type A test. Inspections required by the Maintenance Rule (10 CFR 50.65) and by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code are performed to identify containment degradation that could affect leak tightness.

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.

The proposed extension to the interval for the Type A test does not involve any design or operational changes that could lead to a new or different kind of accident from any accidents previously evaluated. The tests are not being modified, but are only being performed after a longer interval. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation.

Therefore, the proposed change does not create the possibility of a new or