50-416

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 3, 1996

Mr. C. Randy Hutchinson
Vice President, Operations GGNS
Entergy Operations, Inc.
P. O. Box 756
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SUBJECT: CORRECTION TO THE SAFETY EVALUATION FOR AMENDMENT NO. 127 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1 (TAC NO. M95316)

Dear Mr. Hutchinson:

By letter August 21, 1996, we issued Amendment No. 127 to the Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GGNS). The amendment authorized a revision of the schedule for withdrawing capsules with reactor vessel material specimens in accordance with the reactor vessel material surveillance program for GGNS and Section III.B.3 of Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10 CFR Part 50.

In the Safety Evaluation that supported the amendment and was an enclosure to the letter, we overlooked a missing word and an incorrect sentence on pages 2 and 3 of the evaluation. The missing word is "in" and it should have been in the phrase "to calculate the shift 'in' RT_{NDT}", of the first sentence of the second paragraph from the bottom of page 2.

The incorrect sentence, at the bottom of page 2 and top of page 3, was the following: The limiting condition is the pressure test, and the P-T curve is calculated using the conservative lower bound static crack initiation fracture toughness (K_{IC}) (where K_{IC} is approximately 2.4 times K_{IR}). The correct sentence is the following: The limiting condition is the pressure test, and the P-T curve is calculated using the conservative lower bound K_{IR}. K_{IC}, the static crack initiation fracture toughness, is approximately 2.4 times K_{IR}, which demonstrates the conservatism in the P-T limits calculation.

These corrections do not affect the conclusions in the Safety Evaluation and in the letter of August 21, 1996.

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The new corrected pages 2 and 3, containing marginal lines indicating the area of change, are enclosed. The old pages 2 and 3 of the Safety Evaluation should be removed and replaced by the new pages.

Sincerely,

Jack N. Donohew, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosure: Corrected pages 2 and 3 of the Safety Evaluation for Amendment No. 127

cc w/encl: See next page

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cc w/encl: See next page

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The number of GGNS specimen holders was determined per ASTM E185-73 (Ref. 3). The three specimen holders were designed, built and analyzed to Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), 1971 Edition, with Addenda through Winter 1972. Each holder has 12 Charpy V-notch (CVN) specimens of the weld, base metal and heat-affected zone (HAZ) for a total of 36 specimens. A set of unirradiated specimens are kept, as well as archive material, for additional testing in order to provide baseline information.

GGNS is defined as an ASTM E185-73 Case "A" plant since the vessel has a shift in the reference null-ductility temperature (ΔRT_{NDT}) of less than 100°F and will be exposed to a neutron fluence of less than $5x10^{18}$ n/cm^2 over the design lifetime of the plant. The current testing schedule requires that the first specimen holder be removed at 8 EFPY, the second at 24 EFPY, and the testing and reporting is to be performed in accordance with the more recent ASTM E185-82 (Ref. 4). If the ASTM E185-82 requirements were applied to determine the schedule the first capsule should be withdrawn when the vessel wall fluence is $5x10^{18}$ n/cm^2, or when the ΔRT_{NDT} reaches 50°F whichever is first. The GGNS vessel wall is unlikely to reach the conditions described above during the design lifetime of the plant, therefore, early capsule withdrawal is not critical for continued operation of the plant.

In response to Generic Letter (GL) 92-01, Supplement 1 (Ref. 5), a study was performed by General Electric Company for the Boiling Water Reactor (BWR) Vessel and Internals Project (VIP) on the copper levels present in BWR beltline materials (Ref. 6). The purpose was to verify plants with significant variation in the reported copper levels. GGNS was determined to be consistent with reported values with no significant variation in the reactor vessel material (e.g., 0.02-0.06% copper).

The licensee used Regulatory Guide (RG) 1.99, Revision (Rev.) 2 to calculate the shift in RT_{NDT} and adjusted reference temperature (ART) values for all GGNS beltline materials. The fluence used to evaluate the 32 EFPY ART was 2.5 $x10^{18}$ n/cm^2. The resulting predicted values of RT_{NDT} shift indicate that the vessel will not experience a large shift over vessel life. A comparison was made between calculated shift and fluence values and actual surveillance data from other BWR's in order to confirm the conservative predicted shift plus margin values that were used to modify the surveillance program schedule. The results for BWR/6's, including GGNS, show a small shift for capsules removed at EFPY similar to Grand Gulf's current schedule and at higher fluence levels. Based on the data, the measured shift for GGNS would be conservatively bound by the RG 1.99, Rev. 2 calculations.

The shift in RT_{NDT} that results from surveillance testing is used to determine the crack arrest fracture toughness (K_{IR}). K_{IR} is used to calculate the pressure-temperature (P-T) limits curves. The current P-T limits curves for GGNS are calculated with the 10 EFPY shift in RT_{NDT}. The limiting condition is the pressure test, and the P-T curve is calculated using the conservative lower bound K_{IR}. K_{IC}, the static crack initiation fracture toughness, is

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approximately 2.4 times K_{IR}, which demonstrates the conservatism in the P-T limits calculation. Vessel fracture toughness is not a significant concern for GGNS over the life of the vessel.

The BWR Owner's Group (BWROG) began a supplemental surveillance program (SSP) in the late 1980's that was designed to significantly increase the amount of BWR surveillance data in a systematic manner. The BWROG's reasons for beginning the program were the following:

- There are a smaller number of capsules per plant and fewer BWRs than pressurized water reactors (PWRs)
- There are not much BWR surveillance data at higher fluences, nor would there be for many years
- RG 1.99, Rev. 2 imposed some hardships on pressure testing for BWRs, some of which might be relieved if a better understanding of BWR embrittlement phenomenon were obtained.

Supplemental capsules were installed in Cooper and Oyster Creek, and specimen withdrawal is planned for 1996, 2000, and 2002. The results will be the equivalent of 84 additional surveillance capsules compared to approximately 25 which have been tested. The materials used were selected to bound the range of chemistries in BWR beltline materials, and in most cases are BWR beltline materials. The GGNS surveillance plate and weld material, including the limiting material, are among the materials in the capsules. At least one of these materials is in each of the seven capsules in the SSP. Results will be developed which will provide information on all the GGNS plate and weld surveillance materials, and will be directly applicable to the GGNS surveillance program. Specifically, the capsules, when tested, will have collected between $5x10^{17}$ n/cm^2 and $2x10^{18}$ n/cm^2 fluence, which bounds the end of life (EOL) fluence for the GGNS vessel.

Since the expected shift is low, the first surveillance program testing should be at a time when the majority of the shift in the vessel RT_{NDT} has been achieved. Early testing may yield shift in RT_{NDT} values that are not distinguishable from the data scatter. Anomalous shift is not a major concern because, if it were to occur, the BWROG SSP will identify any greater than expected shift.

The staff used RG 1.99 calculational methods to verify that the 8 EFPY fluence value ($2.2\ x10^{17}$ n/cm^2) in combination with the low copper values (0.02-0.06%) result in predicted values that are not distinguishable from the data scatter. The staff also verified that the 24 EFPY fluence value ($6.9\ x10^{17}$ n/cm^2) would result in predicted values that are more likely to be distinguishable from the data scatter.

The licensee determined the revised surveillance schedule by examining the fracture toughness decrease as a function of shift, and used that shift to determine the appropriate EFPY for removal and testing of the first capsule.