Entergy

Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150
Tel 601 437-6470

February 11, 1998

W. K. Hughey
Director
Nuclear Safety & Regulatory Affairs

. S. Nuclear Regulatory Commission
Mail Station P1-37
Washington, D.C 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-29
Request for Alternative to 50.55a Examination Requirements of Category B1.11 Reactor Vessel Welds, Relief Request I-2-00001

GNRO-98/00015

Gentlemen:

In 1995 the Boiling Water Reactor Vessel & Internals Project (BWRVIP) transmitted BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations" to the Nuclear Regulatory Commission (NRC). The NRC's evaluation prompted numerous industry/NRC meetings and the issuance of NRC Information Notice 97-63.

NRC issued IN 97-63 to inform addressees of the Status of NRC staff's review of "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)." Because of additional analysis requested by the NRC, the staff indicated in the information notice that consideration will be given to technically-justified requests for relief from the augmented examination in accordance with 10 CFR 50.55a(a)(3)(i), 10 CFR 50.55a(a)(3)(ii), and 10 CFR 50.55a(g)(6)(ii)A(5) for BWR licensees who are scheduled to perform inspections of the BWR RPV cicumferential shell welds during the Fall 1997 or Spring 1998 outage seasons. Further, acceptably-justified reliefs would be considered for inspection delays of up to two operating cycles for BWR RPV circumferential shell welds only.

Grand Gulf Nuclear Station (GGNS) hereby requests schedular relief from the afformentioned requirements. The attached relief request has been prepared in accordance with the guidance of IN 97-63 using the BWRVIP suggested format. This alternative is being requested under the provisions of 10 CFR 50.55a(a)(3)(i) and will provide an acceptable level of quality and safety. The Inservice Inspection Program at GGNS complies with the 1992 edition with portions of the 1993 addenda of the ASME Boiler and Pressure Vessel Code, Section XI.

A047/1

GGNS requests your review and approval by April 3, 1998 to support GGNS's refueling outage scheduled to commence April 11, 1998. Thank you in advance for your prompt attention in this matter. Should you have any questions or need any additional information, please contact Bill Brice at 601-437-6556.

Yours truly,

WKH/WBB

attachment: Relief Request I-2-00001

cc: Ms. J. L. Dixon-Herrity, GGNS Senior Resident (w/a)
Mr. L. J. Smith (Wise Carter) (w/a)
Mr. N. S. Reynolds (w/a)
Mr. H. L. Thomas (w/o)

Mr. E. W. Merschoff (w/a)
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Mr. J. N. Donohew, Project Manager (w/2)
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop 13H3
Washington, D.C. 20555

ENTERGY OPERATIONS, INC.
GRAND GULF NUCLEAR STATION
2 nd TEN YEAR INTERVAL
REQUEST NO. I-2-00001

I. COMPONENT / EXAMINATION IDENTIFICATION:

Code Class:	1
References:	ASME Section XI, 1992 Edition IWB-2500; 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(g)(6)(ii)(A)(2); BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection"; NRC Information Notice 97-63, "Status of NRC Staff's Review of BWRVIP-05"; BWRVIP Response to NRC RAI on BWRVIP-05, 12/22/97
Examination Category:	B-A
Item No.:	B1.11
Examination Required:	Volumetric Examination of Welds and Adjacent Base Materials
Description:	Circumferential Shell Welds in Reactor Vessel
Component Number:	Q1B13D001

II. REQUIREMENTS:

ASME Section XI, 1992 Edition, IWB-2500 requires the subject welds and associated base material to be volumetrically examined once each interval. The examinations are to be dispersed over the three periods of the interval within the limits specified by IWB-2412-1. Deferral of the examinations until the end of the interval is permissible; however, the examinations during the initial interval were not deferred, and IWB-2420 requires the sequence of examinations established in the first interval to be repeated during subsequent intervals to the extent practical.

In 1992, Title 10 of the Code of Federal Regulations (10 CFR) were amended with the addition of 50.55a(g)(6)(ii)(A), "Augmented Examination of Reactor Vessel." Section 50.55a(g)(6)(ii)(A)(2) requires licensees to augment their reactor vessel examinations by implementing once, as part of the inservice inspection interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel" in Table IWB-2500-1 of subsection IWB of the 1989 Edition of ASME Section XI, subject to the conditions specified in 50.55a(g)(6)(ii)(A)(3) and (4). The augmented examination when not deferred in accordance with the provisions of 50.55a(g)(6)(ii)(A)(3), shall be performed in accordance with the related procedures specified in the Section XI Edition and Addenda applicable to the inservice inspection interval in effect on September 8, 1992. For the purpose of this augmented examination, "essentially 100%", as used in Table IWB-2500-1, means more than 90% of the examination volume of each weld, where the reduction in coverage is due to interference by another component or part geometry.

Section 50.55a(g)(6)(ii)(A)(3) permits licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992, to defer the augmented reactor vessel examination specified in 50.55a(g)(6)(ii)(A)(2) to the first period of the next inspection interval under certain conditions. However, if the augmented examinations are deferred to the first period of the next inspection interval, 50.55a(g)(6)(ii)(A)(3)(vi) requires the deferred examinations to be performed in accordance with the related procedures specified in the Section XI edition and addenda applicable to the inspection interval in which the augmented examination is performed.

Section 50.55a(g)(6)(ii)(A)(4) indicates that the requirement for augmented examination of the reactor vessel may be satisfied by an examination of essentially 100% of the reactor shell welds specified in 50.55a(g)(6)(ii)(A)(2) that have been completed, or are scheduled for implementation with a written commitment, or are required by 50.55a(g)(4)(i), during the inservice inspection interval in effect on September 8, 1992.

III. BASIS FOR ALTERNATIVE

Pursuant to the provisions of 10 CFR 50.55a(a)(3)(i), and consistent with information contained in NRC Information Notice 97-63, an alternative is requested from the examination of RPV circumferential welds as required by ASME Section XI, IWB-2500, Examination Category B-A, Item No. B1.11, and 10 CFR 50.55a(g)(6)(ii)(A)(2) as described within. This proposed alternative is to postpone the examination of the RPV circumferential welds for two operating cycles, until Refueling Outage RF11 that is presently scheduled to begin in approximately April, 2001. The basis for this request for alternative is documented in the report "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)" that was transmitted to the NRC in September 1995 and BWRVIP Response to NRC RAI on BWRVIP-05 that was transmitted to the NRC on December 18, 1997.

The BWRVIP-05 report provides the technical basis for eliminating examinations of BWR RPV circumferential shell welds. The BWRVIP-05 report concludes that the probability of failure of the BWR RPV circumferential shell welds is orders of magnitude lower than that of the axial shell welds. Additionally, the NRC assessment demonstrated that examination of BWR RPV circumferential shell welds does not measurably affect the probability of failure. Therefore the NRC evaluation appears to support the conclusions of BWRVIP-05.

The independent NRC assessment utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate RPV failure probabilities. Three key assumptions in the PFM analysis are:

(1) the neutron fluence was that estimated to be end-of-license (EOL) mean fluence,
(2) the chemistry values are mean values based on vessel types, and
(3) the potential for beyond-design-basis events is considered.

Although BWRVIP-05 provides the technical basis supporting the request for alternative, the following information is provided to show the conservatisms of the NRC analysis for the Grand Gulf Nuclear Station (GGNS). For plants with RPVs fabricated by Chicago Bridge & Iron (CB&I), the mean EOL neutron fluence used in the NRC PFM analysis was 0.19E+19 n/cm^2. However, the highest surface fluence for the GGNS RPV beltline region at the end of the requested alternative period is predicted to be 0.102E+19 n/ cm^2. Thus the effect of fluence on embrittlement is much lower, and the NRC analysis as described in the NRC independent assessment is conservative for GGNS in this regard. Therefore, there is significant conservatism in the already low circumferential-weld-failure probabilities as related to GGNS. Other GGNS RPV shell weld information that the NRC staff has requested be included in this relief request is provided in the attached Table 1.

As shown in UFSAR Figure 5.3-9, GGNS does not have any circumferential welds in the beltline region. However, an evaluation showing the effects of radiation have been performed on the two circumferential welds that are closest to the core. The effects of irradiation depicted in this relief request are significantly exaggerated because:

- the two welds are not located in the peak fluence region of the beltline, however peak beltline fluence values have been used in their evaluation (weld AB is approximately 5 inches below the core and weld AC is approximately 22 inches above the core),

- the fluence used in this relief request represents surface fluence and not 1/4t fluence, and

- there is no credit taken for the attenuation caused by the RPV inner surface cladding.

The results of the evaluations are listed in Table 1. As shown in Table 1, the calculated embrittlement shift in RT_{NDT} (i.e., ΔRT_{NDT}) for the GGNS Unit 1 vessel is a maximum of 22.76°F at the end of the requested relief period. By comparison, using the mean values for fluence and weld chemistry assumed for CB&I reactor vessels in Table 7-5 of Enclosure 1 to the NRC independent assessment report, a ΔRT_{NDT} of 30.16°F would be derived. Therefore, the calculated ΔRT_{NDT} value for the GGNS vessel is less than, and thus bounded by, the embrittlement shift assumed in the NRC's independent assessment. Furthermore, it can be seen in the attached Table 1 that the calculated Upper Bound RT_{NDT} value for the GGNS near-beltline welds is a maximum of 25.52°F at the end of the requested relief period. For comparison, the Upper Bound RT_{NDT} value in Table 7-8 of Enclosure 1 to the NRC's independent assessment report of BWRVIP-05 is 32.7°F for fluence reference case 1. Again, the calculated Upper Bound RT_{NDT} values for the GGNS vessel circumferential welds are clearly bounded by the limiting RT_{NDT} from Table 7-8 (CB&I vessels) of the NRC independent assessment report, thus providing additional assurance that the GGNS vessel welds are also bounded by BWRVIP-05 report.

An added safety margin has been provided at GGNS by the nondestructive examination (NDE) of the vessel welds. A complete Preservice Inspection (PSI) was performed on all of the RPV shell welds, both longitudinal and circumferential, to the maximum extent practical before GGNS initially loaded fuel. The same welds have also completed Inservice Inspection (ISI) ultrasonic examinations required during the first 10-year interval. The examination coverage for both PSI and ISI for all welds except for circumferential weld AA exceeded 90% coverage of the full volume. Weld AA has been examined over its complete length, but due to scanning limitations from the lower head side of the weld, it was only examined for approximately 67% of the Code required volume.

At the August 8, 1997 meeting and in the NRC's independent assessment, the NRC staff indicated that the potential for, and consequences of, nondesign-basis events not addressed in the BWRVIP-05 report should be considered. In particular, the NRC staff stated that nondesign-basis cold over-pressure transients should be considered. It is highly unlikely that a BWR would experience a cold over-pressure transient. In fact, for a BWR to experience such an event would generally require several operator errors. At the August 8, 1997 meeting, the NRC staff described several types of events that could be precursors to BWR RPV cold over-pressure transients. These were identified as precursors because no cold over-pressure event has occurred at a U.S. BWR. Also at the August 8 meeting, the NRC staff identified one actual cold over-pressure event that occurred during shutdown at a non-U. S. BWR. This event apparently included several operator errors that resulted in a maximum RPV pressure of 1150 psi with a temperature range of 79°F to 88°F. As a result of the NRC's concerns, the BWRVIP has included in Attachment 1 to their response to the NRC's RAI on BWRVIP-05 significant discussion regarding BWR cold pressurization events. GGNS has reviewed the BWRVIP's response and concurs that the conditions and events are accurately depicted and that the procedures and personnel training at GGNS are comparable to those described by the BWRVIP and are adequate to prevent a cold over-pressure transient event. Consequently, the probability of a cold over-pressure transient is considered to be less than or equal to that used in the NRC analysis described in the NRC independent assessment and is conservative for GGNS.

IV. CONCLUSION

Based on BWRVIP-05, the risk-informed independent assessment performed by the NRC staff, the BWRVIP's response to the NRC's RAI, and the discussion contained within, an alternative to the cited requirements under the provisions of 10 CFR 50.55a(a)(3)(i) to delay the described examinations until Refueling Outage RF11 is reasonable and will provide an acceptable level of quality and safety.

Table 1

GGNS RPV Shell Weld Information

Variable	Variable Value by Weld Seam Identification		
Weld Seam[1]/ Welding Process	AB (Lower Circ. Seam) SAW	AC (Upper Circ. Seam) SAW	AC (Upper Circ. Seam) SMAW
Fluence[2] @ 13.1 EFPY (end of relief request period)	0.102×10^{19} n/cm^2	0.102×10^{19} n/cm^2	0.102×10^{19} n/cm^2
Initial RT_{NDT}	- 40.0 ° F	- 20.0 ° F	- 60.0 ° F
Weld Chemistry Factor	41	54	27
Weld Copper Content	0.03 wt%	0.04 wt%	0.02 wt%
Weld Nickel Content	0.81 wt%	0.95 wt%	0.91 wt%
Increase in Reference Temperature due to Irradiation (ΔRT_{NDT})	17.28 ° F	22.76 ° F	11.38 ° F
Margin Term	17.28 ° F	22.76 ° F	11.38 ° F
Mean Adjusted Reference Temperature. (Mean ART)	- 22.72 ° F	2.76 ° F	- 48.62 ° F
Upper Bound Adjusted Reference Temperature (Upper Bound ART)	- 5.44 ° F	25.52 ° F	-37.24 ° F

NOTES:-

1) GGNS RPV beltline does not contain circumferential welds. Figure 5.3-9 of the GGNS UFSAR shows that weld seam AB is approximately 5 inches below the core and weld seam AC is approximately 22 inches above the core.

2) The value is the peak fluence in the beltline region that was linearly interpolated to 13.1 EFPY. The use of peak fluence from the beltline region to compute the shift due to irradiation of the circumferential welds (that are not actually in the beltline region) provides a conservative upper bound ART_{NDT}.