April 11, 2001

Mr. William A. Eaton
Vice President, Operations GGNS
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION, UNIT 1 - REQUEST FOR ALTERNATIVE TO SECTION 50.55A OF TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS* (10 CFR) FOR EXAMINATION REQUIREMENTS OF CATEGORY B1.11 REACTOR VESSEL CIRCUMFERENTIAL WELDS (TAC NO. MA9787)

Dear Mr. Eaton:

By letter dated July 27, 2000, as supplemented by letter dated March 19, 2001, you submitted Relief Request I-2-00001, Revision 1, concerning the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI (ASME XI) requirements for the Grand Gulf Nuclear Station, Unit 1, Inservice Inspection (ISI) Program. You requested approval for the use of an alternative from the examination of circumferential shell welds on the reactor pressure vessel (RPV) applicable for the remaining term of operation under the initial facility operating license NPF-29. These ISI examinations are required by ASME XI, IWB-2500, Examination Category B-A, Item No. B1.11, and by the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2).

The proposed alternative eliminates the required examination of RPV circumferential shell welds, and retains the requirement for examination of longitudinal (axial) welds in the RPV shell. Volumetric examination of the axial RPV shell welds (ASME XI, IWB-2500, Examination Category B-A, Item No. B1.12) shall be performed for 100% of these welds. Examination of the axial welds shall also include those portions of the circumferential welds that intersect the axial welds.

The alternative was proposed pursuant to the provisions of 10 CFR 50.55a(a)(3)(i), and is consistent with the staff's Safety Evaluation Report of the Boiling Water Reactor Vessel and Internals Project (BWRVIP)-05 report issued July 28, 1998, and the guidance provided by Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998. We have reviewed your request, and, based on the information provided, we conclude that the alternative you have proposed will provide an acceptable level of quality and safety for the remaining term of facility operating license NPF-29. Therefore, the proposed alternative is authorized pursuant to10 CFR 50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i).

The staff's detailed technical review and conclusions are documented in the enclosed safety evaluation.

If you have any questions related to this issue, please contact me at 301-415-2623.

Sincerely,

/RA/

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosure: Safety Evaluation

cc w/encl: See next page

If you have any questions related to this issue, please contact me at 301-415-2623.

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/RA/

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE FOR EXAMINATION REQUIREMENTS OF CATEGORY B1.11 REACTOR VESSEL CIRCUMFERENTIAL WELDS

RELIEF REQUEST I-2-00001, REVISION 1

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated July 27, 2000, as supplemented by letter dated March 19, 2001, Entergy Operations, Inc., licensee for the Grand Gulf Nuclear Station, Unit 1 (GGNS), requested that the Nuclear Regulatory Commission (NRC) approve an alternative to performing circumferential shell weld examinations on the reactor pressure vessel (RPV) welds. These examinations are required by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and by the augmented examination requirements of Title 10 of the *Code of Federal Regulations* Section 50.55a(g)(6)(ii)(A)(2) (10 CFR 50.55a(g)(6)(ii)(A)(2)). The alternative was proposed pursuant to the provisions of 10 CFR 50.55a(a)(3)(i) and is consistent with the staff's Safety Evaluation Report (SER) of the Boiling Water Reactor Vessel and Internals Project (BWRVIP)-05 Report issued July 28, 1998, and the guidance provided in Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief From Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998.

1.1 Regulatory Requirements

Pursuant to the requirements of 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components are to meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

The rules at 10 CFR 50.55a(g)(6)(ii)(A) require that licensees perform an augmented RPV shell weld examination as specified in the 1989 Edition of Section XI of the ASME Code. The final Rule was published in the *Federal Register* on August 6, 1992 (57 FR 34666). By incorporating

into the regulations the 1989 Edition of the ASME Code, the NRC staff required that licensees perform volumetric examinations of "essentially 100 percent" of the RPV pressure-retaining shell weld, during all inspection intervals. Pursuant to 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph (g) may be used when authorized by the NRC if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

1.2 BWRVIP-05 Report

By letter dated September 28, 1995, as modified and supplemented by letters dated June 24 and October 29, 1996, and May 16, June 4, June 13, and December 18, 1997, the BWRVIP submitted the proprietary report BWRVIP-05, "BWR [Boiling Water Reactor] Vessel and Internals Project, BWR Reactor Vessel Shell Weld Inspection Recommendations." As modified, the BWRVIP report proposed to reduce the scope of inspection of BWR RPV welds from essentially 100 percent of all RPV shell welds to examination of essentially 100 percent of the axial (i.e., longitudinal) welds, and essentially zero percent of the circumferential RPV shell welds, except at the intersection of the axial and circumferential welds, thereby including approximately two to three percent of the circumferential welds. In addition, the report provided proposals to revise ASME Code requirements for successive and additional examinations of circumferential welds, provided in paragraph IWB-2420(b) of Section XI of the ASME Code.

On July 28, 1998, the NRC staff issued an SER of the BWRVIP-05 report. This evaluation concluded that the failure frequency of RPV circumferential welds in BWRs was sufficiently low to justify elimination of inservice inspection (ISI) of these welds. In addition, the SER concluded that the BWRVIP proposals on successive and additional examinations of circumferential welds were acceptable. The SER indicated that examination of the circumferential welds shall be performed if axial weld examinations reveal an active, mechanistic mode of degradation.

In the BWRVIP-05 report, the BWRVIP concluded that the conditional probabilities of failure for BWR RPV circumferential welds are orders of magnitude lower than that of the longitudinal welds. As a part of its review of the report, the NRC conducted an independent risk-informed, probabilistic fracture mechanics assessment of the results presented in the BWRVIP-05 report. The staff assessment conservatively calculated the conditional probability of failure from RPV axial and circumferential welds during the (current) initial 40-year license period and at conditions approximating an 80-year vessel lifetime for a BWR nuclear plant, as indicated in Tables 2.6-4 and 2.6-5 of the SER, respectively. The failure frequency for an RPV is calculated as the product of the frequency for the critical (limiting) transient event and the conditional probability of failure for the weld.

The staff determined the conditional probability of failure for longitudinal and circumferential welds in BWR vessels fabricated by Chicago Bridge and Iron (CB&I), Combustion Engineering, and Babcock and Wilcox. The analysis identified a cold over-pressure event in a foreign reactor as the limiting event for BWR RPVs, with the pressure and temperature from this event used in the probabilistic fracture mechanics calculations. The staff estimated that the probability for the occurrence of the limiting over-pressurization transient was 1×10^{-3} per reactor year. For each of the vessel fabricators, Table 2.6-4 of the staff’s evaluation identifies the conditional failure probabilities for the plant-specific conditions with the highest projected reference temperature (for that fabricator) after the initial 40-year license period.

1.3 Generic Letter 98-05

On November 10, 1998, the NRC issued Generic Letter (GL) 98-05 "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds." GL 98-05 stated that BWR licensees may request permanent (i.e., for the remaining term of operation under the existing, initial, license) relief from the ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential RPV welds (ASME Code Section XI, Table IWB-2500-I, Examination Category B-A, Item 1.11, "Circumferential Shell Welds"), upon demonstrating that:

(1) at the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 30, 1998, safety evaluation, and

(2) licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 30, 1998, safety evaluation.

Licensees would still need to perform the required inspections of "essentially 100 percent" of all axial welds.

2.0 INFORMATION PROVIDED BY THE LICENSEE

This section describes the Code requirements and the components for which the licensee is seeking relief, the basis for the relief request, and a demonstration by the licensee that the criteria for relief are satisfied.

2.1 Code Requirements For Which Relief Is Sought

The licensee identifies the following Code requirements from which relief is sought:

(1) ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, volumetric examination of welds and adjacent base materials. Permanent relief (i.e., for the remaining term of operation under the existing license) is requested.

2.1.1 Components for Which Relief Is Sought

The requested permanent relief from the Table IWB-2500-1 requirements applies to:

ISI Class 1, Code Category B-A, "Pressure Retaining Welds in Reactor Vessel," item B1.11, "Circumferential Shell Welds"

2.2 Licensee's Evaluation of Materials

The licensee's request is based upon provisions in the NRC SER for the BWRVIP-05 report and the guidance outlined in GL 98-05. These documents provide the basis for the elimination of ISIs of BWR RPV circumferential shell welds.

As described previously, GL 98-05 provides two criteria that relief request applicants must demonstrate. One criterion is based upon the limiting conditional failure probability of the applicant's circumferential welds. The other criterion is based upon the implementation of operator training and establishment of procedures to limit the frequency of cold over-pressure events. These criteria are intended to demonstrate that the conditions at the applicant's plant are bounded by those in the SER.

The NRC SER for the BWRVIP-05 report evaluated the conditional failure probability of circumferential welds for the limiting plant-specific case of BWR RPVs manufactured by different vendors, including CB&I, using the highest mean irradiated RT_{NDT} to determine the limiting case.

Since the GGNS RPV was fabricated by CB&I, the relief request compared the mean RT_{NDT} at 32 effective full power years (EFPYs) for GGNS to that for the limiting CB&I case described in Table 2.6-4 of the Final SER of the BWRVIP-05 report. As illustrated in Table 1, the mean RT_{NDT} for GGNS is lower than that for the limiting CB&I case, and the licensee concluded that the conditional failure probability for the GGNS circumferential welds is bounded by the conditional failure probabilities in the staff's SER through the end of the current license period.

Table 1: Comparison of GGNS Circumferential Weld and the NRC Limiting Plant Specific Analysis from Table 2.6-4 of the Final SER of the BWRVIP-05 Report

PARAMETER	GGNS COMPARATIVE DATA AT 32 EFPY (BOUNDING CIRCUMFERENTIAL WELD-HEAT 5P6771)	USNRC LIMITING PLANT SPECIFIC ANALYSIS DATA AT 32 EFPY (CB&I)
Fluence (10^{19} n/cm^2)	0.250	0.51
Initial RT_{NDT} (°F)	-20	-65
Chemistry Factor (°F)	54	109.5
Cu (Wt. %)	0.04	0.10
Ni (Wt. %)	0.95	0.99
ΔRT_{NDT} (°F)	33.69	109.5
Mean RT_{NDT} (°F) [Initial RT_{NDT} + ΔRT_{NDT}]	13.69	44.5

2.3 Licensee's Evaluation of Procedures and Potential Injection Sources

During review of the BWRVIP-05 report, the staff identified non-design basis events which should have been considered in the BWRVIP-05 report. In particular, the potential for and consequences of cold over-pressure transients should be considered. The licensee has

assessed the systems that could lead to a cold over-pressurization of the GGNS RPV. These include the reactor core isolation cooling (RCIC), high pressure core spray (HPCS), low pressure core spray (LPCS), low pressure core injection (LPCI), standby liquid control (SBLC), condensate and feedwater, control rod drive (CRD), and reactor water cleanup (RWCU) systems.

The RCIC system is one of the GGNS high pressure makeup systems. The RCIC system is driven by a steam turbine. During cold shutdown conditions, there is no steam available for operation of the system. Therefore, the RCIC cannot contribute to an over-pressurization event during cold shutdown. The other high pressure makeup system is the HPCS system. The HPCS system is motor driven. The HPCS injection valve is closed on reactor vessel high-water level to prevent overfilling. This high-water level interlock is only overridden for testing purposes, however, injection is prevented during the testing evolution by either racking out the pump breaker or closing the manual injection valve. Therefore, it is unlikely that inadvertent HPCS initiation would result in an over-pressurization event.

The LPCS system is a low pressure emergency core cooling system (ECCS) spray system. The LPCS system discharge pressure is about 500 psig. The technical specification (TS) pressure-temperature (P/T) limits permit pressures up to about 300 psig at temperatures from 70 °F to 100 °F. At temperatures over 100 °F, the permitted pressure increases immediately to above 700 psig. Plant procedures specify that temperature normally be maintained between 120 °F and 130 °F. During refueling outages, there is typically a short period of time during vessel head detensioning and following vessel head retensioning when the temperature would be less than 100 °F. In the event of inadvertent LPCS actuation during these conditions, instrumentation and alarms would be available to the operators pertaining to the LPCS system and reactor level conditions. Procedures contain a cautionary statement that directs operators upon an inadvertent LPCS actuation, to immediately evaluate adequate core cooling and secure the LPCS pump. The licensee concluded that the procedural controls and short period of time when the vessel temperatures could be below 100 °F, make the probability for a over-pressurization due to inadvertent LPCS actuation very low.

The LPCI system is a low pressure ECCS injection system. The LPCI discharge pressure is about 300 psig. With the reactor metal temperature of the vessel maintained above 70 °F, as required by the TSs, the discharge pressure of the LPCI system is not sufficient to exceed the TS P/T limit.

There are no automatic starts associated with the SBLC system. SBLC injection requires operator action to manually start the system by a key lock switch. The SBLC is a low flow rate system (about 42 gpm per pump) and there is a limited supply of water contained in the SBLC storage tank (about 5,000 gallons). In the unlikely event of inadvertent manual SBLC initiation, there would not be enough water in the storage tank to result in pressurization of the reactor.

The reactor feed pumps are the high pressure makeup system during normal operation. The feed pumps are steam driven and cannot be operated during cold shutdown. In addition, the feedwater injection valves are closed once the plant has entered cold shutdown with a reactor coolant temperature around 200 °F. Therefore, the feed pumps cannot contribute to an over-pressurization event during cold shutdown.

The condensate pumps are motor driven and have a discharge pressure of about 200 psig. This pressure is not sufficient to exceed the reactor P/T limits. For the P/T limits to be exceeded, both a condensate pump and a condensate booster pump would have to be inadvertently manually initiated and manually lined up for injection through the feedwater injection valves, which would have to be opened. The operators would have numerous indications of condensate system injection prior to the reactor being pressurized above the P/T limits. The licensee concluded that this scenario was of very low probability.

During normal cold shutdown conditions, RPV level and pressure are normally controlled through a feed and bleed process using the CRD and RWCU systems. The operators closely monitor reactor water level, temperature, and pressure during cold shutdown conditions. The CRD system flow rate, about 60 gpm, is low enough that operators should have sufficient time to respond to any unexpected changes. The CRD and RWCU systems are also used to maintain level during primary system hydrostatic testing. Strict controls, such as limiting reactor pressure changes to 50 psi per minute and requiring two safety relief valves to be operable during the test, minimize the likelihood of an over-pressure event during a hydrostatic test.

Operators are trained in methods of controlling water level within specified limits, in addition to responding to abnormal water level conditions during shutdown. Procedures and controls for reactor temperature, level, and pressure are in place to minimize the potential for RPV cold over-pressurization events. Plant-specific procedures have been established to provide guidance to the operators regarding compliance with the TS P/T limits.

2.4 Licensee's Proposed Alternative Examination

Pursuant to 10 CFR 50.55a(a)(3) licensees may propose alternatives to the requirements of 10 CFR 50.55a(g). The licensee proposed, as an alterative, to perform longitudinal (axial) weld examinations and incidental examination of two to three percent of the intersecting circumferential shell welds to the maximum extent possible based on accessibility. The licensee would permanently defer examination of the circumferential welds until expiration of the plant's current operating license.

3.0 NRC STAFF'S EVALUATION

The staff's review focused on confirming that the licensee has adequately documented that the conditions for relief outlined in the SER to the BWRVIP-05 report and GL 98-05 are satisfied.

3.1 Relaxation From Circumferential Weld Examination Requirements

3.1.1 Circumferential Weld Conditional Failure Probability

In the case of GGNS, there are no circumferential welds within the beltline region. As such, circumferential welds AB and AC could be considered to be the limiting welds which are located five inches below the bottom of the active fuel region and 22 inches above the top of the active fuel region, respectively. The corresponding fluence values (E > 1.0 MeV) were assumed to be the peak value calculated at the end-of-license within the active fuel region. In addition no credit was taken for the stainless steel vessel cladding. The end-of-license fluence value is predicted to be: 0.25×10^{19} n/cm^2 for both AB and AC welds.

The staff's SER provides a limiting conditional failure probability of 2 x 10^{-7} per reactor year for a limiting plant-specific mean RT_{NDT} of 44.5 °F for CB&I-fabricated RPVs. Comparing the information submitted in the relief request, the staff has confirmed that the mean RT_{NDT} of the circumferential welds at GGNS is projected to be 13.69 °F at the end of the current license. In this evaluation, the chemistry factor, ΔRT_{NDT}, and mean RT_{NDT} were calculated consistent with the guidelines of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The calculated value of mean RT_{NDT} for the circumferential welds at GGNS is significantly lower than that for the limiting plant-specific case for CB&I-fabricated RPVs, indicating that the conditional failure probability of the GGNS circumferential welds is much less than 2 x 10^{-7} per reactor year.

3.1.2 Cold Overpressure Transient Probability

On the basis of the evaluation of high pressure injection sources, operator training, and established plant-specific procedures, the licensee determined that appropriate controls are in place to minimize the potential for RPV cold over-pressurization events. The information provided regarding the GGNS high pressure injection systems, operator training, and plant-specific procedures, provides a sufficient basis to support approval of the alternative examination request. The staff concludes that a non-design basis cold over-pressure transient is unlikely to occur at GGNS.

4.0 CONCLUSION

The staff has reviewed the licensee's submittal and finds that the licensee has provided an acceptable demonstration that the appropriate criteria in GL 98-05 and the staff's SER of the BWRVIP-05 report have been satisfied regarding permanent relief (i.e., for the remaining term of operation under the initial, existing license) from ISI requirements for the volumetric examination of RPV circumferential welds, ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11.

The NRC staff concludes that authorization of the licensee's alternative examinations would provide assurance of structural integrity and, therefore, an acceptable level of quality and safety. Accordingly, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative examination for GGNS is authorized.

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