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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO ALTERNATIVE INSPECTION OF REACTOR PRESSURE VESSEL

CIRCUMFERENTIAL WELDS

ENTERGY OPERATIONS, INC., ET. AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated February 11, 1998, Entergy Operations, Inc. (the licensee) requested an alternative to performing the reactor pressure vessel (RPV) circumferential shell weld examinations requirements of both (1) the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 1992 Edition, with portions of the 1993 Addenda (inservice inspection), and (2) the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2), for the Grand Gulf Nuclear Station, Unit 1 (GGNS). The alternative was proposed pursuant to the provisions of 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(g)(ii)(A)(5), and is consistent with information contained in Information Notice (IN) 97-63, "Status of NRC Staff Review of BWRVIP-05," dated August 7, 1997.

The licensee will perform the inspections of essentially 100 percent of the RPV shell longitudinal seam welds and essentially zero percent of the RPV shell circumferential seam welds, which will result in partial examination (2 to 3 percent) of the circumferential welds at or near the intersections of the longitudinal and circumferential welds. This would be done in accordance with the BWRVIP-05 and the ASME Code requirements (i.e., one-third of the welds inspected each 40 months of the current 10-year interval).

Inservice inspections at GGNS, which include the RPV circumferential weld inspection, shall be performed in accordance with Section XI of the ASME Code, Class 1, 2, and 3 components, and applicable addenda, as required by 10 CFR 50.55a(g) of the regulations. Pursuant to the requirements of 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components shall 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 the ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months

ENCLOSURE

prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable ASME Code, Section XI, for GGNS, during the current 10-year ISI interval is the 1992 Edition, with portions of the 1993 Addenda.

Section 50.55a(g)(6)(ii)(A) to Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a(g)(6)(ii)(A)) requires that licensees perform an expanded RPV shell weld examination as specified in the 1989 Edition of Section XI of the ASME Code, on an "expedited" basis. "Expedited," in this context, effectively meant during the inspection interval when the Rule was approved or the first period of the next inspection interval. 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 examination of "essentially 100 percent" of the RPV pressure-retaining shell welds during all inspection intervals. Section 50.55a(a)(3)(i) (10 CFR 50.55a(a)(3)(i)) indicates that alternatives to the requirements in 10 CFR 50.55a(g) are justified when the proposed alternative provides an acceptable level of quality and safety.

By letter dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, and May 16, June 4, June 13 and December 18, 1997, the Boiling Water Reactor Vessel and Internals Project (BWRVIP), a technical committee of the BWR Owners Group (BWROG), submitted the proprietary report, "BWR Vessel and Internals Project, BWR Reactor Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," which proposed to reduce the scope of inspection of the BWR RPV welds from essentially 100 percent of all RPV shell welds to 50 percent of the axial welds and 0 percent of the circumferential welds. By letter dated October 29, 1996, the BWRVIP modified their proposal to increase the examination of the axial welds to 100 percent from 50 percent while still proposing to inspect essentially 0 percent of the circumferential RPV shell welds, except that the intersection of the axial and circumferential welds would have included approximately 2 to 3 percent of the circumferential welds.

On May 12, 1997, the NRC staff and members of the BWRVIP met with the Commission to discuss the NRC staff's review of the BWRVIP-05 report. In accordance with guidance provided by the Commission in Staff Requirements Memorandum (SRM) M970512B, dated May 30, 1997, the staff has initiated a broader, risk-informed review of the BWRVIP-05 proposal.

In IN 97-63, the staff indicated that it would consider technically-justified alternatives to the augmented examination in accordance with 10 CFR 50.55a(a)(3)(i), 10 CFR 50.55a(a)(3)(ii), and 50.55a(g)(6)(ii)(A)(5), from BWR licensees who are scheduled to perform inspections of the BWR RPV circumferential welds during the fall 1997 or spring 1998 outage seasons. Acceptably-justified alternatives would be considered for inspection delays of up to 40-months or two operating cycles (whichever is longer) for BWR RPV circumferential shell welds only.

2.0 BACKGROUND - STAFF ASSESSMENT OF BWRVIP-05 REPORT

The staff's independent assessment of the BWRVIP-05 proposal is documented in a letter dated August 14, 1997, to Mr. Carl Terry, BWRVIP Chairman. The staff concluded that the industry's assessment did not sufficiently address risk, and additional work is necessary to provide a complete risk-informed evaluation.

The staff's assessment was performed for BWR RPVs fabricated by Chicago Bridge and Iron (CB&I), Combustion Engineering (CE), and Babcock & Wilcox (B&W). The staff assessment identified cold over-pressure events as the limiting transients that could lead to failure of BWR RPVs. Using the pressure and temperature resulting from a cold over-pressure event in a foreign reactor and the parameters identified in Table 7-1 of the staff's independent assessment, the staff determined the conditional probability of failure for axial and circumferential welds fabricated by CB&I, CE, and B&W. Table 7-9 of the staff's assessment identifies the conditional probability of failure for the reference cases and the 95 percent confidence uncertainty bound cases for axial and circumferential welds fabricated by CB&I, CE and B&W. B&W fabricated vessels were determined to have the highest conditional probability of failure. The input material parameters used in the analysis of the reference case for B&W fabricated vessels resulted in a reference temperature (RT_{NDT}) at the vessel inner surface of 114.5°F. In the uncertainty analysis, the neutron fluence evaluation had the greatest RT_{NDT} value (145°F) at the inner surface. Vessels with RT_{NDT} values less than those resulting from the staff's assessment will have less embrittlement than the vessels simulated in the staff's assessment and should have a conditional probability of vessel failure less than or equal to the values in the staff's assessment.

The failure probability for a weld is the product of the critical event frequency and the conditional probability of the weld failure for that event. Using the event frequency for a cold over-pressure event and the conditional probability of vessel failure for CB&I fabricated circumferential welds, the best-estimate failure frequency from the staff's assessment is $<6.0 \times 10^{-11}$ (1) per reactor year and the uncertainty bound failure frequency is $<2.8 \times 10^{-10}$ (1) per reactor year.

3.0 ALTERNATIVE

The alternative proposed by the licensee is to postpone beginning the examination of the RPV circumferential welds until the 11th refueling outage at GGNS. This period would be about 3 years from the plant shutdown for the 9th refueling outage in April 1998. The examination of the RPV circumferential welds would then have to be completed by the end of the current 10-year interval.

4.0 LICENSEE'S TECHNICAL JUSTIFICATION

The licensee indicated in its February 11, 1998, letter that the basis for requesting the alternative inspections is the BWRVIP-05 report, which stated that the probability of failure of BWR RPV circumferential shell welds is orders of magnitude lower than that of the axial shell welds. This conclusion was also demonstrated in the staff's independent assessment of the BWRVIP-05 report. The BWRVIP-05 report indicates that, for a typical BWR RPV, the failure probability for axial welds is 2.7×10^{-7} and the failure probability for circumferential welds is 2.2×10^{-11} for 40 years of plant operation.

The licensee calculated the RT_{NDT} value for the limiting GGNS circumferential welds at the end of the requested relief period using the methodology in Regulatory Guide (RG) 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Revision 2. Since there are no circumferential welds in the beltline region, the limiting circumferential welds are weld seams AB and AC which are approximately 5 inches below and 22 inches above the reactor core, respectively. The RT_{NDT} values were calculated in accordance with RG 1.99, Revision 2, and depend upon the neutron fluence, the amounts of copper and nickel in the circumferential weld, and its unirradiated RT_{NDT}. The licensee determined the highest neutron

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(1) Insufficient or no failures to accurately determine reference case failure probability.

fluence at the end of the next two operating cycles at the inner surface of the circumferential welds to be 0.102 x 10^{19} n/cm^2 . This value resulted from linearly interpolating the peak fluence in the beltline region to the end of the relief request period. The amounts of copper and nickel in weld seam AB are 0.03 percent and 0.81 percent, respectively. The amounts of copper and nickel in weld seam AC are 0.04 percent and 0.95 percent, respectively. The plant -specific unirradiated RT_{NDT} for weld seam AB is -40°F, and the plant-specific unirradiated RT_{NDT} for weld seam AC is -20°F. Using these parameters and the methodology in Regulatory Guide 1.99, Revision 2, the licensee determined that the RT_{NDT} values for circumferential weld seams AB and AC at the end of the relief period are -5.4°F and 25.5°F, respectively. Both values are less than the reference case for the CB&I fabricated vessels in the staff's assessment. Since the RT_{NDT} values of the GGNS circumferential welds are less than the values in the staff's independent assessment, the licensee concluded that the GGNS vessel circumferential welds are bounded by the staff's independent assessment, thus providing additional assurance that the vessel welds are also bounded by the BWRVIP-05 report.

The licensee assessed the systems that could lead to a cold overpressurization of the RPV. These included the high pressure core spray (HPCS), reactor core isolation cooling (RCIC), standby liquid control (SBLC), reactor feed pumps, condensate system, low pressure core spray (LPCS), low pressure core injection (LPCI), control rod drive (CRD) and reactor water cleanup systems (RWCU).

The RCIC pumps are steam driven and do not function during cold shutdown. The licensee stated that the RCIC turbine was designed to operate on Auxiliary Steam for testing purposes, however, operation with Auxiliary Steam is not allowed by procedure. The HPCS system is motor operated and could be operated during cold shutdown. Startup of HPCS requires either manual initiation or inadvertent initiation. The HPCS system has a high level interlock for the HPCS injection valve to prevent overfilling the RPV. This interlock cannot be overridden and therefore, overpressurization of the RPV should not occur. The licensee stated that there were no automatic starts associated with SBLC. Operator initiation of SBLC should not occur during shutdown, however, the SBLC injection rate is approximately 42 gpm which would allow operators sufficient time to control reactor pressure if manual initiation occurred.

The reactor feed pumps are the high pressure makeup system during normal operations. The reactor feed pumps are steam driven and therefore, cannot be operated during cold shutdown. The condensate system is the supply source to the reactor feed pumps. The condensate pumps require manual initiation and line up for injection during cold shutdown. If the resulting reactor level increase was ignored, the reactor pressure-temperature limit would not be exceeded due to the condensate shutoff head of approximately 150 psig. The LPCS and LPCI systems are low pressure ECCS systems with low shutoff heads. If either one of these systems were manually or inadvertently initiated during cold shutdown, the resulting reactor pressure and temperature would be below the pressure-temperature limits. The CRD and RWCU systems use a feed and bleed process to control RPV level and pressure during normal cold shutdown conditions. Per plant procedures, the reactor head vents are open when reactor coolant temperature is less than 190°F. The CRD pumps injection rate is less than 60 gpm; this flowrate and the opened reactor head vents allow sufficient time for operators to react to unanticipated level changes.

In all cases, the operators are trained in methods of controlling water level within specified limits in addition to responding to abnormal water level conditions during shutdown. The licensee also stated that procedural controls for reactor temperature, level, and pressure are an integral part of operator training. Plant-specific procedures have been established to provide guidance to the operators regarding compliance with the Technical Specification pressure-temperature limits. On the basis of the pressure limits of the operating systems, operator training, and established plant-specific procedures, the licensee determined that a non-design basis cold over-pressure transient is unlikely to occur during the requested delay. Therefore, the licensee concluded that the probability of a cold over-pressure transient is considered to be less than or equal to that used in the staff's assessment of BWRVIP-05.

5.0 RPV NEUTRON FLUENCE

The licensee based the relief request on the values of the fluence submitted in the attachment GE-NE-B1301807-01R1 to its letter of May 2, 1996. The letter of May 2, 1996, addressed a change in the schedule for the removal of the surveillance capsules which was approved in the staff's letter of August 21, 1996.

The attachment reported on the results of dosimeter measurements and the resulting estimate of the end of life (32 effective full power years (EFPYs) of operation). The peak fluence value in the beltline region was estimated to be 0.25x10+19 n/cm2. The proposed delay for the inspection is that it be performed at 13.1 EFPYs. The prorated peak fluence value at this time is 0.102x10+19 n/cm2. Dosimetry measurements if they are not accompanied by benchmarked calculations may not be very accurate. However, in this case the RPV does not have beltline welds. The two welds to be inspected are 5 inches below and 22 inches above the active core respectively. The fast neutron fluence decreases significantly with distance away from the active core. Therefore, whatever uncertainty is encountered in the measured estimation of the beltline fluence is adequately compensated by fluence reduction due to their distance from the active core.

In addition the licensee stated that the estimated fluence was for the vessel cladding rather than the vessel surface, which constitutes another conservatism. Finally the staff notes that the end of life estimated ΔRT_{NDT} for the end of life of the plant is about 28°F. The total ΔRT_{NDT} at the time of the postponed inspection will be with in the uncertainty band of the shift measurement. Therefore, the requested relief fulfills the 10 CFR 50.55a requirements.

6.0 STAFF REVIEW OF LICENSEE TECHNICAL JUSTIFICATION

The staff confirmed that the RT_{NDT} values for the circumferential welds at the end of the relief period are less than the values in the reference case and uncertainty analysis for the CB&I fabricated vessels. RT_{NDT} is a measure of the amount of irradiation embrittlement. Since the RT_{NDT} values are less than the values in the reference case and the uncertainty analysis for CB&I fabricated vessels, the GGNS RPV will have less embrittlement than the CB&I fabricated vessels and will have a conditional probability of vessel failure less than or equal to that estimated in the staff's assessment.

The staff reviewed the information provided by the licensee regarding the GGNS high pressure injection systems, operator training, and plant-specific procedures to prevent RPV cold over-pressurization. The information provided sufficient basis to support approval of the alternative examination request. The staff concludes that the probability of a non-design basis cold over-pressure transient occurring at GGNS during the requested delay is low, which is consistent with the staff's assessment.

7.0 CONCLUSIONS

Based upon its review, the staff reached the following conclusions:

1) Based on the licensee's assessment of the materials in the circumferential welds in the GGNS RPV, the conditional probability of vessel failure should be less than or equal to that estimated from the staff's assessment.

2) Based on the licensee's high pressure injection systems analyses, operator training, and plant specific procedures, the probability of a non-design-basis cold over-pressure transient is low during the requested delay period and is consistent with the staff's assessment.

3) Based on the RPV neutron fluence values submitted by the licensee in 1996, whatever uncertainty is encountered in the measured estimation of the beltline fluence is adequately compensated by fluence reduction due to their distance from the active core.

4) Based on the above three conclusions, the staff concludes that the GGNS RPV can be operated during the requested delay period with an acceptable level of quality and safety, and the inspection of the circumferential welds may be delayed for the requested two operating cycles.

Therefore, the proposed postponement of beginning the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2) at Grand Gulf for circumferential shell welds for two operating cycles is authorized pursuant to 10 CFR 50.55a(a)(3)(i), because the alternative to the ASME Code requirements provides an acceptable level of quality and safety at GGNS.

The staff is reviewing, but has not approved, the use of the BWRVIP-05 requirements for this examination. If the staff approves the BWRVIP-05 requirements, you will need to request relief from the ASME Code requirements to use these requirements. If the staff does not approve these requirements, you will need to redo the examinations or request relief from the ASME Code requirements.

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