# AmerGen

AmerGen Energy Company, LLC
200 Exelon Way
Suite 345
Kennett Square, PA 19348

www.exeloncorp.com

An Exelon/British Energy Company

10 CFR 50.55a

September 14, 2001
2130-01-20195

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington DC 20555

Subject: Oyster Creek Generating Station
Docket No. 50-219
ASME Section XI Relief Request R17, Revision 1

Attached is a revised request for relief from requirements contained in ASME Section XI. This relief is requested pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5). This revision is necessary since examination coverage for reactor vessel shell axial welds was less than projected. NRC staff review and approval is requested by June 30, 2002.

Should you have any questions or require any additional information please contact Mr. Paul F. Czaya at 609-971-4139.

Very truly yours,

Michael P. Gallagher

Michael P. Gallagher
Director - Licensing

Enclosure: ASME Section 11 Code Request R17, Revision 1

c: H. J. Miller, Administrator, USNRC Region I
L. A. Dudes, USNRC Senior Resident Inspector, Oyster Creek
H. N. Pastis, USNRC Senior Project Manager, Oyster Creek
File No. 01042

A047

## Relief Request R17, Revision 1
## Request for Relief from Achieving More Than 90% of the Examination Volume of Certain Reactor Pressure Vessel Axial Shell Welds

I. Component for Which Relief is Requested

Reactor Vessel Shell Welds, Category B-A, Item B1.12

Upper Axial Shell Weld (non-beltline): 2-563C, 2-563D, 2-563E, and 2-563F

Lower Axial Shell Weld (includes beltline): 2-564A, 2-564C, 2-564D, 2-564E, and 2-564F

II. Code Requirement

Section XI (1986 Edition), Table IWB-2500-1, Category B-A, Item B1.12 requires examination of all welds in the 1st inspection interval and one beltline region weld in the successive inspection intervals.

10 CFR 50.55a(g)(6)(ii)(A)(2) states that all licensees shall augment their reactor vessel examinations by implementing the examination requirements for Reactor Pressure Vessel (RPV) 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 Section XI, Division 1, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(A)(3) and (4). As stated in 10 CFR 50.55a(g)(6)(ii)(A)(2) for the purposes of this augmented examination, "essentially 100%," as used in Table IWB-2500-1, means more than 90 percent of the examination volume for each weld. Additionally, 10 CFR 50.55a(g)(6)(ii)(A)(5) requires licensees that are unable to completely satisfy the augmented RPV shell weld examination requirement to submit information to the U.S. Nuclear Regulatory Commission to support the determination and propose an alternative to the examination requirements that would provide an acceptable level of quality and safety.

III. Code Requirement from which Relief is Requested

Relief is requested from the ASME Code Section XI requirement to examine essentially 100% (defined in 10 CFR 50.55a(g)(6)(ii)(A)(2) as more than 90%) of the volume of welds identified in Table 1 that do not meet this criterion.

## IV. Basis for Relief

Oyster Creek (OC), a Boiling Water Reactor (BWR-2), was designed and built before ASME Code Section XI was developed and access for inspections became a design requirement. As a result, there is little external access to the outside diameter of reactor pressure vessel (RPV) axial shell welds due to inadequate clearance between the bioshield wall and vessel insulation. Examinations were performed during the 1R18 refueling outage in October/November 2000 from the inside diameter (ID) using the General Electric GERIS-2000 inspection system.

The OC examination plan requires examination of 100% of all accessible regions of the RPV axial shell welds. Code coverage of 90% or greater was obtained for only three welds, as opposed to six welds projected in the Oyster Creek access study described in Reference 1. The three welds where Code coverage was obtained are 2-563A and 2-563B located above the beltline region, and 2-564B within the beltline region. The limitations for the other nine welds were due to the physical constraints of the RPV internal design, arrangement of internal components and the core shroud tie-rod repair as identified in Table 2. In the initial Relief Request R17 (Reference 1) the projected coverage was anticipated based on the RPV accessibility study and clearance measurements for the GERIS-2000 inspection tool. As described in Reference 1, several internal vessel components prevented a 100% ID ultrasonic test examination including interference from the feedwater sparger, vessel material specimen brackets, vibration brackets, shroud conical support plate, and shroud repair tie rod assembly.

Weld examination coverage was lower than expected due to problems encountered with the feedwater sparger and weld locations in the lower intermediate shell course that were not as shown on vessel drawings. Three welds in the lower intermediate shell course (2-564A, B and C) were approximately 6 degrees counter clockwise (CCW) from the location indicated on the drawings. Two welds could not be examined (2-564A and 2-564F) and examination coverage for two welds was significantly lower than projected (2-563C and 2-564C).

Table 1 identifies weld examination coverage. Table 2 identifies examination limitations. Figure 1 shows examination scan areas.

**Table 1 – Weld Examination Coverage**

| Weld ID | Weld Configuration | Weld Length (in) | Projected Coverage | Actual Coverage |
|---|---|---|---|---|
| NR02 2-563A* | Longitudinal, Upper Shell @15° AZ | 132.6 | 100.0% | 100% |
| NR02 2-563B* | Longitudinal, Upper Shell @ 135° AZ | 132.6 | 99.2% | 92.8% |
| NR02 2-563C | Longitudinal, Upper Shell @ 255° AZ | 132.6 | 99.4% | 80.4% |
| NR02 2-563D | Longitudinal, Int. Upper Shell @ 330° AZ | 132.6 | 65.3% | 62.3% |
| NR02 2-563E | Longitudinal, Int. Upper Shell @ 90° AZ | 132.6 | 65.3% | 59.7% |
| NR02 2-563F | Longitudinal, Int. Upper Shell @ 210° AZ | 132.6 | 62.6% | 59.7% |
| NR02 2-564A | Longitudinal, Lower Int. Shell @ 219° AZ | 133.6 | 93.0% | 0 |
| NR02 2-564B* | Longitudinal, Lower Int. Shell @339° AZ | 133.6 | 93.0% | 100% |
| NR02 2-564C | Longitudinal, Lower Int. Shell @99° AZ | 133.6 | 94.1% | 39.7% |
| NR02 2-564D | Longitudinal, Lower Shell @258° AZ | 83.4 | 55.1% | 56.1% |
| NR02 2-564E | Longitudinal, Lower Shell @18° AZ | 131.6 | 76.0% | 74.7% |
| NR02 2-564F | Longitudinal, Lower Shell @138° AZ | 131.6 | 76.0% | 0 |

* Meets Code requirements.

**Table 2 – Examination Limitations**

| Weld ID | Examination Limitations | Coverage |
|---|---|---|
| NR02 2-563A | None | 100% |
| NR02 2-563B | Steam Dryer Support Lug | 92.8% |
| NR02 2-563C | Main Steam Nozzle Plug | 80.4% |
| NR02 2-563D | Feedwater Sparger, Manipulator Scan Limits | 62.3% |
| NR02 2-563E | Feedwater Sparger, Manipulator Scan Limits | 59.7% |
| NR02 2-563F | Feedwater Sparger, Manipulator Scan Limits | 59.7% |
| NR02 2-564A | Feedwater Sparger Bolting and Shroud Repair Tie Rod | 0 |
| NR02 2-564B | None | 100% |
| NR02 2-564C | Shroud Repair Tie Rod | 39.7% |
| NR02 2-564D | Recirculation Outlet Nozzle | 56.1% |
| NR02 2-564E | Shroud Repair Tie Rod | 74.7% |
| NR02 2-564F | Feedwater Sparger Brackets Interference With Scanner | 0 |

Welds 2-563D, 2-563E, 2-563F, 2-564D and 2-564E were subject to examination limitations as shown in Table 2. Examination coverage achieved for these welds does not differ significantly from the Reference 1 estimates as shown in Table 1.

The percent of weld 2-563C examined was 80.4% while expectations were to achieve Code coverage. The unanticipated limitation was due to interferences associated with the "B" main steam nozzle plug assembly.

Weld 2-564A in the lower intermediate course was not located at the azimuth appearing on the drawing. It was located behind a tie rod 6 degrees CCW from the position shown on the drawing. Attempts to scan the weld failed because the tie rod and feedwater nozzle sparger bolting created a limitation that was not anticipated. Due to the feedwater nozzle and sparger connection configuration, the transducer was unable to make contact with the weld below. For a proper inspection the transducer must be flush with the weld. This could not be accomplished due to the interference of the feedwater nozzle sparger bolting.

Weld 2-564C in the lower intermediate course was not located at the azimuth appearing on the drawing. The weld was located behind a tie rod 6 degrees CCW from its position shown on the drawing. In the area of the weld two seismic bumpers are located on the tie rod. The tie rod was not expected to interfere in this weld examination, however, the unexpected interference reduced examination coverage to 39.7%.

Weld 2-564F is located directly below a feedwater nozzle where the nozzle to sparger configuration makes it impossible to position the transducer array flush with the weld.

The total examination coverage for the reactor vessel axial welds is less than the 82 percent estimated in Reference 1. Due to reduced coverage the overall inspection amounted to about 60% of the total length of axial welds. The estimate for the axial welds in the beltline region was 100% exam coverage. The actual coverage is about 57%. The NRC staff acceptance (Reference 2) of the estimated reduction in RPV examination coverage was based upon an approximate 82% total vessel exam and a 100% exam of the beltline region. Since the actual coverage was significantly less, an analysis for beltline (the most potentially susceptible area) axial weld failure probability was performed using the VIPER code developed by Structural Integrity Associates, Inc. This evaluation by Structural Integrity was performed to assess the effect on the probability of fracture due to the 57% inspection performed on the vessel axial welds. The evaluation was based on the methodology presented in EPRI Report, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)." Based on this evaluation, the inspections performed at Oyster Creek were effective in inspecting regions that account for essentially all of the vessel fracture failure risk. This is based on a very small failure probability of $2.5 \times 10^{-12}$/year as a result of 57% beltline examination coverage. Therefore, the proposed alternative provides an acceptable level of quality and safety.

References: 1) Correspondence No. 1940-99-20680 dated December 30, 1999, "ASME XI Relief Requests"
2) NRC Letter dated September 13, 2000, "Safety Evaluation of the Request for Relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI Requirements for the Containment Inservice Inspection Program, Oyster Creek Nuclear Generating Station (TAC No. MA8015)"

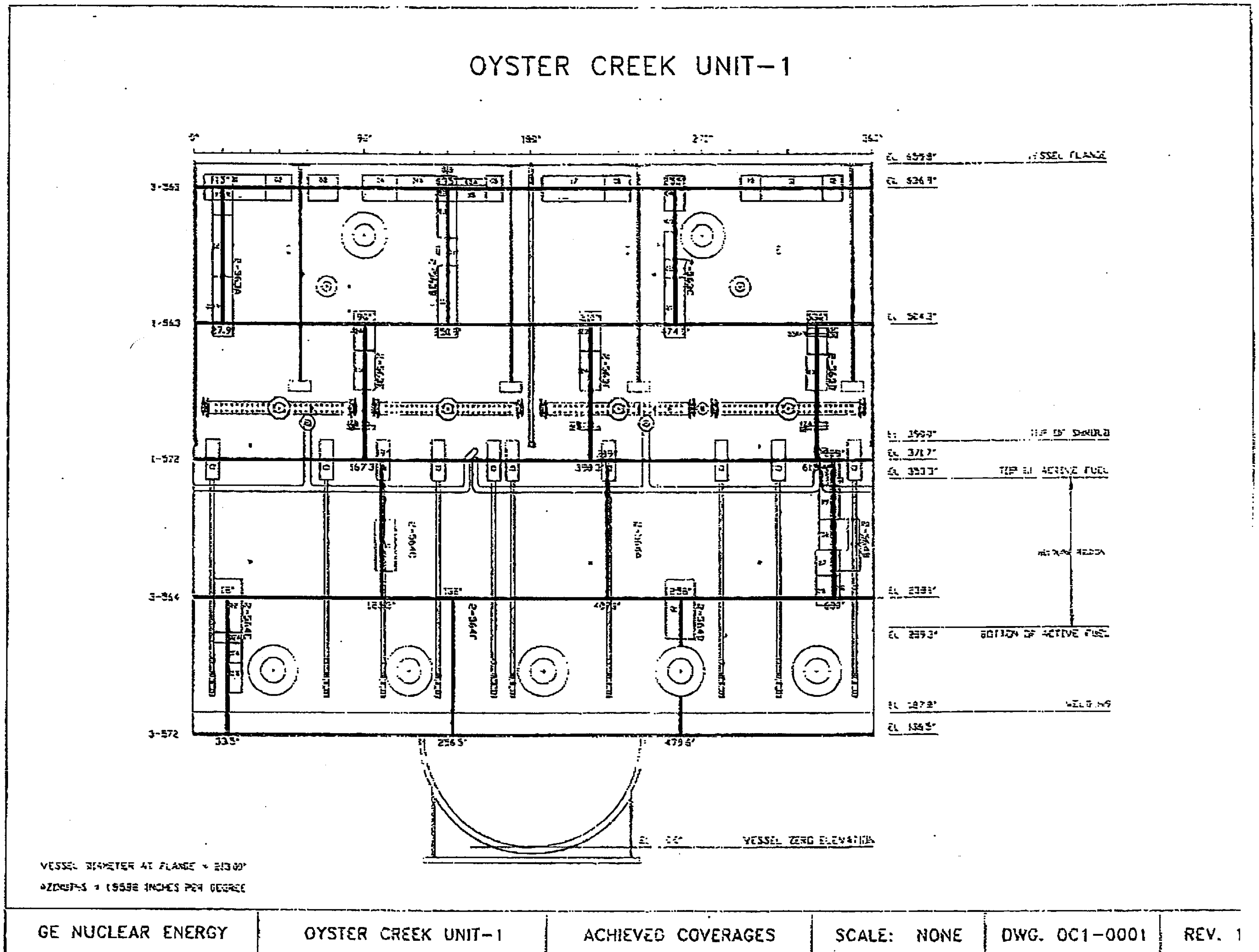


Figure 1 - Examination Scan Areas