GPU NUCLEAR

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December 30, 1999
1940-99-20680

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

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
ASME XI Relief Requests

Attached to this cover letter are two requests for relief from requirements contained in ASME Section XI. This relief is requested pursuant to 10 CFR 50.55(a). To support planning and scheduling for our next refueling outage, staff review and approval is needed by June 30, 2000.

If any additional information or assistance is required, please contact Mr. John Rogers of my staff at 609.971.4893.

Very truly yours,

for
Sander Levin
Acting Director
Oyster Creek

SL/JJR
Attachment

cc: Administrator, Region I
NRC Project Manager
Senior Resident Inspector

A047

Attachment to Letter 1940-99-20680

Oyster Creek Nuclear Generating Station

ASME Relief Requests R17, 18

Relief Request R17

R17 is a request for relief from achieving more than 90% of the examination volume of certain reactor pressure vessel axial shell welds for the remaining term of operation under the existing license and relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of axial reactor vessel (RPV) welds. These projections are based on in-vessel access studies, drawing review, and vessel internals inspection video tape reviews performed by the examination vendor. Actual examination of the axial shell welds may yield a variance from the expected examination coverage. GPU Nuclear has concluded that this alternative provides an acceptable level of quality and safety. Furthermore, compliance with the specified requirements of 10CFR50.55a(g)(6)(ii)(A)(2) would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, this proposed alternative satisfies the requirements of 10CFR50.55a(a)(3)(i) and 10CFR50.55a(a)(3)(ii).

I. Component for Which Relief is Requested

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

Upper Shell (non-beltline): 2-563A, 2-563B, 2-563C, 2-563D, 2-563E, 2-563F

Lower Shell (includes beltline): 2-564A, 2-564B, 2-564C, 2-564D, 2-564E, 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.

However, 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 ASME Boiler and Pressure Vessel Code, subject to the conditions specified in 50.55a(g)(6)(ii)(A)(3) and (4). As stated in 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, 10CFR50.55a(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 Section XI requirement to examine essentially 100% (defined in 50.55a(g)(6)(ii)(A)(2) as more than 90%) of the volume of welds identified in Table 1 with estimated coverages of 90%, or less.

IV. Basis for Relief

Oyster Creek, a BWR-2, was designed and built well before Section XI was developed and access for inspections became a design requirement. As a result, there is little external access to the OD of RPV axial shell welds due to inadequate clearances between the bioshield wall and vessel insulation. Therefore, the examinations are planned to be performed from the ID using the GE GERIS 2000 inspection system.

The OC examination plan requires examination of 100% of all accessible regions of the RPV axial welds. The ability to inspect 100% of the axial welds will be limited, in some cases, due to the physical constraints of the RPV internal vessel design and arrangement of internal components. An internal vessel accessibility study of the RPV was performed by General Electric to determine the inspectability of the RPV axial shell welds and to obtain clearance measurements for the GERIS-2000. Several internal vessel components will limit a 100% ID UT examination including interference from the Feedwater Spargers, Specimen Brackets, Vibration Brackets, the Shroud Support Baffle Plate, and Shroud Repair Tie Rod Assembly.

Table 1 below identifies the weld, the projected coverage, and the physical obstructions that prevent access to each weld. Figures 1 and 2 show a RPV rollout drawing of the vessel welds and access for scanning. Figure 1 shows the coverage obtained with the standard scanning. Figure 2 shows the additional coverage obtained with a slight modification to the scanning package.

Table 1 – Estimated Coverages of OCNGS RPV Axial Welds

Weld ID (See Notes 1 & 2)	Weld Length (in)	Volume Effectively Examined (%)	Volume Examined (%) x Weld Length	Obstruction (See Note 3)
2-563A	132.6	100.0%	132.6	N/A
2-563B	132.6	99.2%	131.5	SDB
2-563C	132.6	99.4%	131.8	IN
2-563D	132.6	65.3%	86.6	FWS, MSL
2-563E	132.6	65.3%	86.6	FWS, MSL
2-563F	132.6	62.6%	83.0	FWS, SB
2-564A	133.6	93.0%	124.2	TR
2-564B	133.6	93.0%	124.2	TR
2-564C	133.6	94.1%	125.7	TR
2-564D	83.4	55.1%	46.0	RON
2-564E	131.6	76.0%	100.0	SSBP/G, TR
2-564F	131.6	76.0%	100.0	SSBP/G, TR
Totals	1543	82.4	1272.2	

NOTES:
1. Welds that are numbered 563 are upper shell welds that are not part of the beltline.
2. Welds that are numered 564 are lower shell welds, parts of which are in the beltline.
3. GR - Guide Rod, SB - Specimen Bracket, CSL - Core Spray Line, FWS - Feedwater Sparger, CSDC – Core Spray Downcomer, IN - Instrumentation Nozzle, MSL - Manipulator Scan Limit, VB - Vibration Bracket, RON – Recirc Outlet Nozzle, SSBP/G - Shroud Support Baffle Plate/Gussets, SDB - Steam Dryer Bracket, TR-Tie Rod

Key conclusions from a review of the information provided in Table 1 and Figures 1 and 2 are:

1. The access to the lower portions of welds 2-563D, E, and F is restricted by the feedwater spargers. Creating access to the lower portion of these welds would involve removal and replacement of the spargers. We consider that this causes an undue hardship and large personnel dose with no concurrent increase in safety.
2. Access to the lower portions of welds 2-564D, E, and F is restricted primarily by the shroud support plate and gussets (slightly above weld H9 as shown on the drawings). These lower portions which cannot be accessed are NOT in the beltline region. We consider the lower portion of these welds to be permanently inaccessible for examination by UT.
3. The coverage of the beltline portions of all the 2-564 welds is well above 90%. And, nearly all the volume of the beltline weld material will be examined.

Based upon our review of the accessibility of these welds and the fact that we will be able to examine essentially 100% of the beltline axial welds, we consider approval of our request for relief will provide adequate safety and quality of the RPV axial weld exams.

Relief Request R18

R18 is a request for permanent relief from performing examinations of reactor pressure vessel circumferential shell welds for the remaining term of operation under the existing license, and relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor vessel (RPV) welds. NRC Generic Letter 98-05, under "Permitted Actions," stated "BWR licensees may request permanent (i.e., for the remaining term of operation under the existing, initial, license) relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds (ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item 1.11, Circumferential Shell Welds) by 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."

Similar relief has been granted for other BWRs, such as Nine Mile Point, Unit 1 which has a vessel nearly identical to OCNGS.

I. Component for Which Relief is Requested

Reactor Vessel Shell Welds, Category B-A, Item B1.11

Weld Numbers: NR02-1-563, NR02-1-572, NR02-3-572, NR02-3-564

II. Code Requirement

Section XI (1986 Edition), Table IWB-2500-1, Category B-A, Item B1.11 requires examination of all welds in the 1st inspection interval and one beltline region weld in the successive inspection intervals.

However, 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 ASME Boiler and Pressure Vessel Code, subject to the conditions specified in 50.55a(g)(6)(ii)(A)(3) and (4). As stated in 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, 10CFR50.55a(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 determine, 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

Permanent (i.e., for the remaining term of operation under the existing license) relief is requested from performing the Section XI requirement and augmented inservice inspection requirement of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor vessel (RPV) welds.

IV. Basis for Relief

On November 10, 1998, the NRC Issued Generic Letter 98-05 (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" (Reference 1) The Generic Letter provided guidance for licensees seeking relief from requirements for examination of BWR RPV circumferential shell welds as recommended in the BWRVIP-05 report (Reference 2). GL 98-05 states that the licensees may request relief from the inservice inspection requirements of 10CFR50.55a(g) for volumetric examination of the RPV circumferential shell welds by demonstrating: (1) at the expiration of their license, the circumferential shell welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 28, 1998, safety evaluation (Reference 3), 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 28, 1998, safety evaluation.

A. This section provides the basis that demonstrates that, at the expiration of the OCNGS license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 28, 1998, safety evaluation (of GL 98-05, Permitted Action).

<u>Adjusted Reference Temperature (ART), Weld Chemistry, and Fluence</u>

In the OCNGS RPV, the limiting weld for estimating conditional failure probability of the circumferential welds is the one circumferential weld in the beltline region. This is weld 3-564, which was manufactured using heat number 1248. In GPUN's response to a Request for Additional Information (RAI) regarding Reactor Pressure Vessel Integrity at Oyster Creek Nuclear Generating Station (OCNGS) Reference (4), the ART for heat number 1248 at 32 EFPY is 77°F. This value is well below the Mean RT_{NDT} value of 98.1°F used by the NRC in its limiting probabilistic fracture mechanics analysis, as identified in the staff's SE, Table 2.6-4, of BWRVIP-05 (Ref. 3). Additionally, OCNGS will have operated for about 29.1 EFPY at the end of the current license; this is less than the 32 EFPY operation assumed in the fluence estimate (i.e., another conservatism). The following table is provided for comparison purposes.

Parameter	OCNGS Limiting Circumferential Weld, Heat 1248,	USNRC LIMITING PLANT SPECIFIC ANALYSIS PARAMETERS AT 32 EFPY SE (Ref. 3) TABLE 2.6-4	
	(Best Estimates) Ref. 4	SE - VIP	SE –Combustion Engineering Owner Group
Fluence (ID), 10^{19} n/cm^2	0.374	0.2	0.2
Fluence Factor	0.728	0.569	0.569
Initial RT_{NDT}, °F	-50	0	0
Cu, %	0.21	0.13	0.183
Ni, %	0.07	0.71	0.704
Chemistry Factor	98	151.7	172.2
ΔRT_{NDT}, °F	71	86.4	98.1
ART, °F	77	86.4	98.1

Since the ART for the OCNGS limiting circumferential weld is less than the mean RT_{NDT} value used by the staff for the limiting weld reference case for the CEOG case, the OCNGS limiting circumferential weld has less embrittlement than the corresponding weld in the NRC limiting weld. Therefore, we can conclude that the limiting OCNGS RPV circumferential weld has a conditional probability of failure less than or equal to that calculated for the case study.

Therefore, we conclude that we have demonstrated that, at the expiration of the OCNGS license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 28, 1998, safety evaluation, as required by item (1) of the "Permitted Action" section of GL 98-05.

B. This section provides the basis that demonstrates that, at OCNGS, we 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 28, 1998, safety evaluation.

Review of Potential High Pressure Injection Sources:
The sources of high-pressure injection at OCNGS are: the Condensate/Feedwater, Control Rod Drive (CRD), and Standby Liquid Control (SLC) systems. Oyster Creek has no high-pressure core injection system.

During normal cold shutdown conditions, the high pressure injection system status is as follows:

1. The Feedwater pumps are secured and one condensate pump is left operating on minimum recirculation flow. The heater string outlet valves are closed and the feedwater regulating valves and flow regulating valves are in the manual mode and closed. The one operating condensate pump (shutoff head of approximately 350 psig) maintains the system full, allows condensate flow through the in-service demineralizer and can be used as a source of make-up to the RPV. It would require several operator errors to inadvertently start a feedwater pump and inject into the vessel.

2. RPV level and pressure are normally controlled with the CRD and Reactor Water Cleanup (RWCU) systems using a "feed and bleed" process. Plant procedures require the establishment of a vent path after the reactor has been cooled to less than 212°F. If either of these systems were to fail, the Operator would adjust the other system to control level. Under these conditions, the CRD system typically injects water into the reactor at a rate of <60 gpm. This slow injection rate allows the operator sufficient time to react to unanticipated level changes and, thus, maintains a low probability of a violation of the pressure-temperature limits.

   Assuming a normal shutdown RPV level of 185 inches in TAF, there are approximately 4200 cubic feet of volume of free space in the RPV. This corresponds to approximataly 31,400 gallons of water. At an injection rate of 60 gpm, assuming a loss of RWCU letdown and no flow through the established vent path, the operator has in excess of 500 minutes to secure CRD and stop the inventory increase.

3. The SLC system is normally in standby. There are no automatic starts associated with this system. SLC injection requires an Operator to manually start the system from the Control Room using a key lock controller.

Additionally, the injection rate of the SLC pump is approximately 30 gpm, which would give the Operator ample time to control reactor pressure in the case of an inadvertent injection as described with the use of the CRD system. The only time the vessel is taken solid is during the performance of NSSS leak testing at the end of each refueling outage.

Pressure testing of the RPV is classified as an "Infrequently Performed Test or Evolution" which ensures that these tests receive special management oversight and procedural controls to maintain the plant's level of safety within acceptable limits. The pressure test is conducted so that the Tech Spec-required temperature bands for the pressure increases are achieved and maintained prior to increasing pressure. During performance of an RPV pressure test, level and pressure are controlled using the CRD and RWCU systems using a "feed and bleed" process. The rate of pressure increase is limited to <50 psig per minute. This practice minimizes the likelihood of exceeding the pressure-temperature limits during performance of the test. Additionally overpressure protection is provided by three electromatic relief valves set at 1085 psig and four of the nine safety valves with a setpoint of 1212 psig.

Procedural Controls/Operator Training to Prevent Reactor Pressure Vessel Cold Over-Pressurization

Operating procedure restrictions, operator training, and work control processes at OCNGS provide appropriate controls to minimize the potential for RPV cold over-pressurization events.

During cold shutdown conditions, reactor water level, pressure, and temperature are maintained within established bands in accordance with operating procedures. Control Room Operators periodically monitor indications and alarms to detect abnormalities as early as possible. Shift supervision is contacted during any abnormal condition. Therefore, any deviations in reactor water level or temperature from a specified band will be promptly identified and corrected. Finally, plant conditions and ongoing activities that could affect critical plant parameters are discussed at each shift turnover. This ensures that on-coming Operators are cognizant of activities that could adversely affect reactor level, pressure, and temperature.

Procedural controls for reactor temperature, level, and pressure are an integral part of Operator training. Specifically, Operators are trained in methods of controlling water level within specified limits, as well as responding to abnormal water level conditions outside the established limits. Plant-specific procedures have been developed to provide guidance to the Operators regarding compliance with the Technical Specification requirements on pressure-temperature limits.

During plant outages, the work control processes assure that the outage schedule and changes to the schedule receive a thorough shutdown risk assessment review to ensure defense-in-depth is maintained. Work activities are reviewed by Station management and Operations management to ensure safe operation and that the plant mode can support the scheduled work.

During outages, work is coordinated through the Outage Command Center and the Ops Work Center which provides an additional level of Operations oversight. In the Control Room, the Shift Manager is required, by procedure, to maintain cognizance of any activity that could potentially affect reactor level or decay heat removal during refueling outages. The Control Room Operators are required to provide positive control of reactor water level within the specified bands, and promptly report when operating outside the specified band, including restoration actions being taken.

Pre-job briefings are conducted for complex work activities, such as RPV pressure tests or hydrostatic testing that have the potential of affecting critical RPV parameters. Pre-job briefings are attended by cognizant individuals involved in the work activity. Expected plant responses and contingency actions to address unexpected conditions, or responses that may be encountered, are included in the briefing discussion.

Based upon the above, the probability of a low temperature over-pressure event at OCNGS is considered to be less than or equal to that used in the USNRC safety evaluation.

Conclusion

Deferral of the RPV circumferential shell weld exams to the end of the current operating license ensure a high degree of quality and safety. Based on the documentation in the BWRVIP-05 report, the risk-informed independent assessment performed by the NRC staff, the lower neutron fluence, the less challenging design and operational loading for BWRs, the quality of the original vessel fabrication, the lack of significant degradation mechanisms, and controls to prevent a cold over-pressure event, GPUN considers that a permanent deferral from performing the augmented inspections of the RPV circumferential shell welds to the end of the current operating license of the OCNGS provides, an acceptable level of quality and safety.

V. ALTERNATIVE PROVISIONS

In accordance with 10CFR50.55a(a)(3)(I) and 10CFR50.55a(g)(6)(ii)(A)(5), GPUN proposes the following alternative provisions for the subject weld examinations.

A. The alternative to the Code- and regulatory-required examinations of the circumferential welds is the incidental examination of circumferential welds (about 2-3%) where they intersect the axial welds, which will be examined as required by Table IWB-2500-1 of the Code and 10CFR50.55a(g). This alternative is as noted in the "Permitted Action" section of GL 98-05.

B. Successive Examination of Flaws

For ASME Code Section XI, Table IWB-2500- 1, Examination Category B-A, Item No. B1.11, "Reactor Pressure Vessel Shell Circumferential Welds," at intersections with longitudinal (axial) welds, successive examinations per IWB-2420, "Successive Inspections," are not required for non-threatening flaws (i.e., original vessel material or fabrication flaws, such as inclusions, which exhibit negligible or no growth during the design life of the vessel), provided that the following conditions are met:

1. The flaw is characterized as subsurface in accordance with BWRVIP-05.
2. The NDE technique and evaluation that detected and characterized the flaw as originating from material manufacture or vessel fabrication is documented in a flaw evaluation report; and,

3. The vessel containing the flaw is acceptable for continued service in accordance with IWB-3600, "Analytical Evaluation of Flaws," and the flaw is demonstrated acceptable for the intended service life of the vessel.

For ASME Code Section XI, Table IWB-2500- 1, Examination Category B-A, Item No. B1.12, "Reactor Pressure Vessel Shell Longitudinal (Axial) Welds," all flaws shall be re-inspected at successive intervals consistent with the ASME Code and regulatory requirements.

C. Additional Examinations of Flaws

For ASME Section X1, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, "Reactor Pressure Vessel Shell Circumferential Welds," at the intersection with longitudinal (axial) welds, additional examination requirements per IWB-2430, "Additional Examinations," are not required for flaws provided the following conditions are met:

1. If the flaw is characterized as sub-surface in accordance with BWRVIP-05, no additional examinations are required;

2. If the flaw is not characterized as subsurface in accordance with BWRVIP-05, an engineering evaluation shall be performed, addressing the following as a minimum:
   - A determination of the root cause of the flaw,
   - An evaluation of any potential failure mechanisms,
   - An evaluation of service conditions which could cause subsequent failure,
   - An evaluation per IWB-3600 demonstrating that the vessel is acceptable for continued service; and

3. If the flaw meets the criteria of IWB-3600 for the intended service life of the vessel, additional examinations may be limited to those welds subject to the root cause conditions and failure mechanisms, up to the number of examinations required by IWB-2430(a). If the engineering evaluation determines that there are no additional welds subject to the same root cause conditions, or no failure mechanism exists, no additional examinations are required.

D. For ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1. 12, "Longitudinal (axial) Reactor Pressure Vessel Shell Welds," additional examination for flaws shall be in accordance with IWB-2430, and successive examinations will be in accordance with IWB-2420. All flaws in RPV shell longitudinal welds shall require additional examinations consistent with the ASME Code and regulatory requirements. Examinations of the RPV shell circumferential welds shall be performed if RPV shell longitudinal (axial) welds reveal an active, mechanistic mode of degradation.

<u>REFERENCES</u>

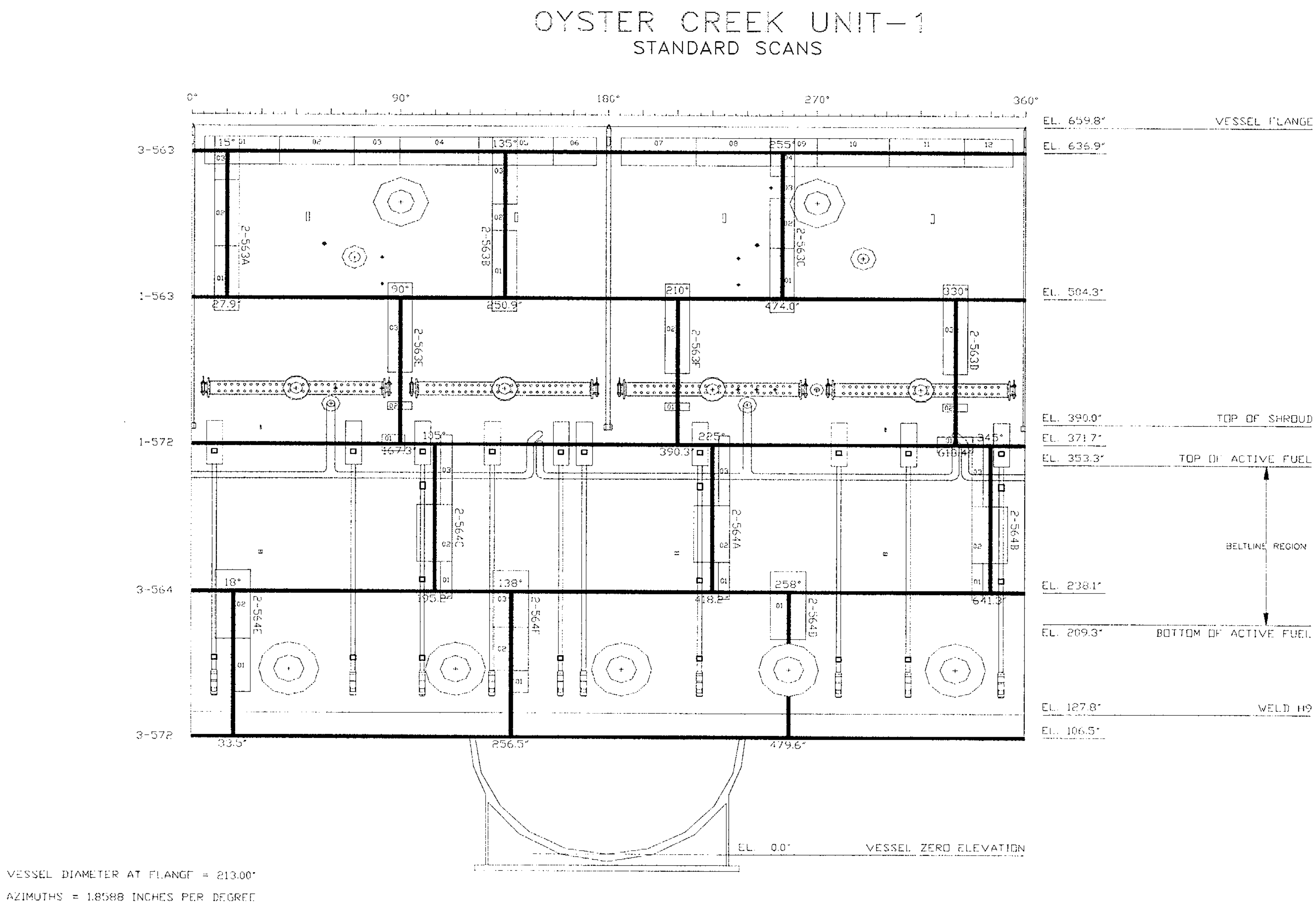
1. NRC Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief From Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998.

2. EPRI TR-105697, BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), September 1995.

3. NRC Letter from Gus C. Lainas, Acting Director, Division Of Engineering, Office Of Nuclear Reactor Regulation, to Carl Terry, BWRVIP Chairman, Niagara Mohawk Company, "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report, July 28, 1998.

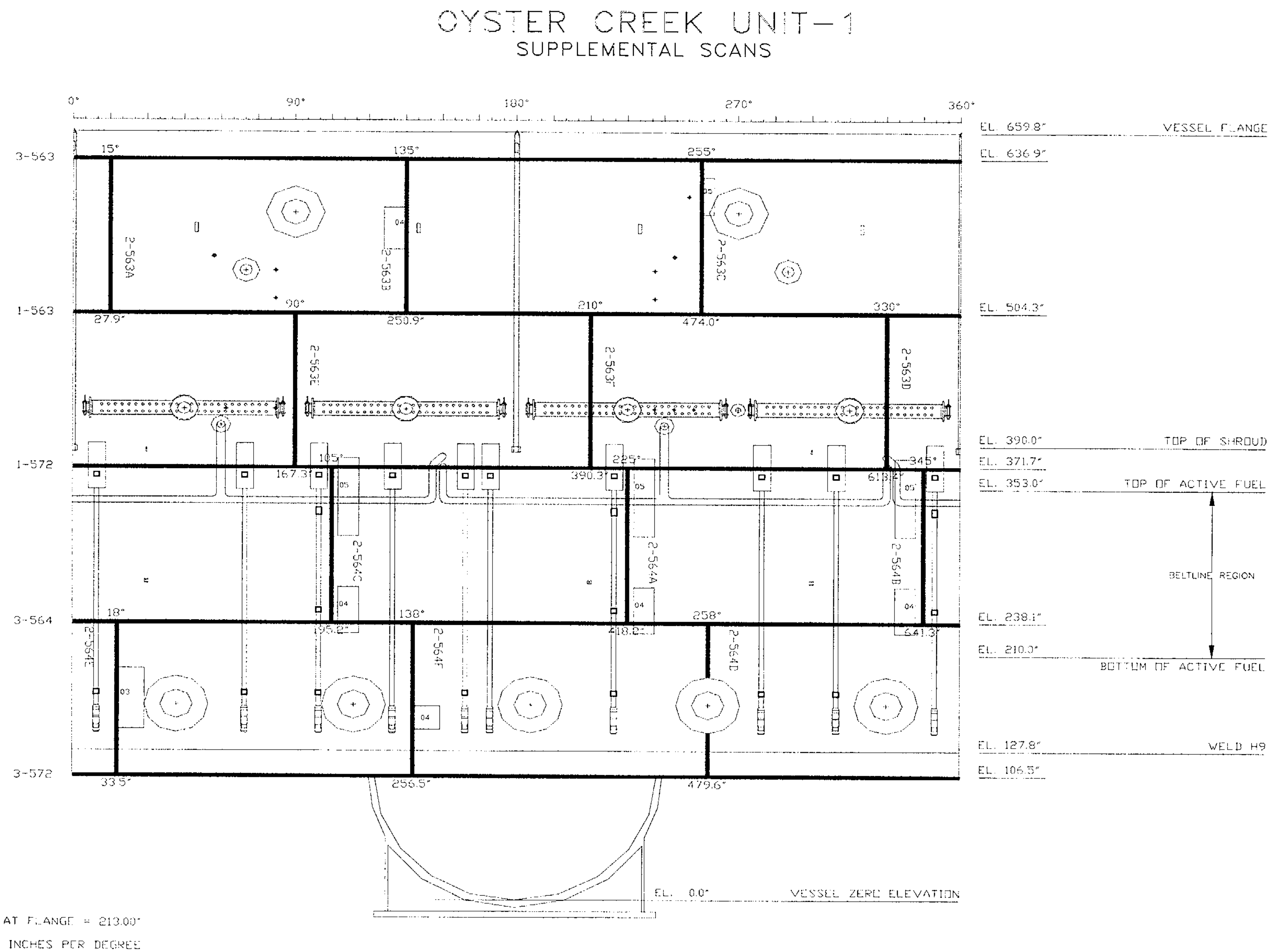
4. GPUN Letter 1940-98-20489, dated September 10, 1998, Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity at Oyster Creek Nuclear Generating Station (OCNGS).

Figure 1



GE NUCLEAR ENERGY	OYSTER CREEK UNIT-1	PROJECTED COVERAGES	SCALE: NONE	DWG. OC1-0001	REV. 1

Figure 2



GE NUCLEAR ENERGY	OYSTER CREEK UNIT-1	PROJECTED COVERAGES	SCALE: NONE	DWG. OC1-0001	REV. 2