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February 27, 2001

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
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Subject: Entergy Operations, Inc.
Request for Alternative to 10CFR50.55a Examination Requirements of Category B1.11 Reactor Pressure Vessel Welds

Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29

CNRO-2001-00006

On July 27, 2000, Entergy Operations, Inc., (Entergy) submitted ASME Relief Request I-2-00001, Rev. 1 for its Grand Gulf Nuclear Station.[1] By this submittal, Entergy requested that the NRC authorize permanent relief from the inservice inspection requirements of 10CFR50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds in accordance with the guidance provided in 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."

In a recent telephone conference call, the NRC staff requested Entergy provide further information pertaining to preventing a cold water overpressure event caused by an inadvertent start of the Low Pressure Core Spray (LPCS) system. Attached is amended request I-2-00001, Rev. 1 providing the requested information. In addition to these changes, we have updated the discussion of the condensate system and have replaced site-specific titles with generic functional titles. Using generic titles will eliminate the need for future revisions caused by position title changes. Revision bars in the page margins denote text changes.

[1] Letter No. CNRO-2000-00021, dated July 27, 2000, "Request for Alternative to 10CFR50.55a Examination Requirements of Category B1.11 Reactor Pressure Vessel Welds"

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This letter contains no commitments.

Should you have any questions regarding this submittal, please contact Guy Davant at (601) 368-5756.

Very truly yours,

M. A Krupa

MAK/GHD/baa
attachments

cc: Mr. W. A. Eaton (GGNS)
Mr. G. R. Taylor (ECH)

Mr. T. L. Hoeg, NRC Senior Resident Inspector (GGNS)
Mr. E. W. Merschoff, NRC Region IV Regional Administrator
Mr. S. P. Sekerak, NRC Project Manager (GGNS)

ENTERGY OPERATIONS, INC.
GRAND GULF NUCLEAR STATION
2nd TEN YEAR INTERVAL
REQUEST NO. I-2-00001, Revision 1

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"; BWRVIP Response to NRC RAI on BWRVIP-05, 12/22/97 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"; NRC Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC NO. M93925), Dated July 28, 1998
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) was 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 Generic Letter 98-05, 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 for the remaining term of operation under the existing license. 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 Safety Evaluation dated July 28, 1998 demonstrates that examination of BWR RPV circumferential shell welds does not measurably affect the probability of failure. Therefore, the NRC evaluation supports the conclusions of BWRVIP-05.

The NRC evaluation of BWRVIP-05 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 plant-specific analysis as they apply to the Grand Gulf Nuclear Station (GGNS):

(1) As defined in ASTM E-185-73, GGNS is a Case "A" plant because the vessel has a predicted shift in the reference nil-ductility temperature (ΔRT_{NDT}) of less than 100°F and will be exposed to a neutron fluence of less than 5E+18 n/cm^2 over the design life of the plant.

(2) For RPVs fabricated by Chicago Bridge & Iron (CB&I), the EOL neutron fluence for the circumferential weld used in the NRC plant-specific analysis is 0.510E+19 n/cm^2 (Reference NRC Safety Evaluation Table 2.6-4). Whereas, the highest surface fluence for the GGNS RPV beltline region at the end of life (32 EFPY) is predicted to be 0.250E+19 n/cm^2. Thus the effect of fluence on embrittlement is much lower, and the NRC analysis as described in the NRC safety evaluation is conservative for GGNS.

(3) The chemistry factor used in the NRC safety evaluation is 109.5 (Reference NRC Safety Evaluation Table 2.6-4) and the chemistry factor for the limiting circumferential weld that is adjacent to the RPV beltline area at GGNS is 54.

(4) 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 has 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.

As described above there is significant conservatism in the already low circumferential-weld-failure probabilities as related to GGNS. Other GGNS RPV shell weld information that compares to the information used in the NRC plant-specific analysis is provided in Table 1.

As shown in Table 1 for the limiting circumferential weld adjacent to the beltline area, the calculated shift in RT_{NDT} (i.e., ΔRT_{NDT}) is a maximum of 33.69°F and the Mean Adjusted Reference Temperature (i.e., ART) is 13.69°F at the end of 32 EFPY. Whereas the values used by the NRC in the plant-specific analysis for the ΔRT_{NDT} is 109.5°F with a resulting ART of 44.5°F. Therefore it is clearly evident that the values used by the NRC in their plant-specific analysis are bounding and provide additional assurance that the GGNS vessel welds are also bound by the BWRVIP-05 report.

An added safety margin is also 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.

In previous evaluations by the NRC, the staff concluded that beyond design basis events occurring during plant shutdown could lead to cold over pressure events that could challenge vessel integrity. As indicated by the NRC safety evaluation, each licensee requesting relief is to demonstrate the implementation of operator training and the existence of procedures that limit the frequency of cold over pressure events to the amount specified in the NRC plant-specific evaluation.

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 plant-specific analysis described in the NRC safety evaluation and is conservative for GGNS.

Review of Potential Injection Sources That Could Cause a Reactor Pressure Vessel Cold Over-Pressurization:

The Reactor Core Isolation Cooling (RCIC) system is one of the high pressure make-up systems at GGNS. The RCIC system is a steam turbine driven system. RCIC injection during cold shutdown is not possible as no steam is available to drive the RCIC turbine. The RCIC turbine was designed to also operate on Auxiliary Steam for testing purposes. The supply line has a removable spool piece and is blind flanged. Operation with Auxiliary Steam is not allowed by procedure.

The High Pressure Core Spray (HPCS) system is another high pressure make-up system at GGNS. The HPCS pump is motor operated so it can be operated when the reactor is in cold shutdown. However, to start the HPCS system would require either manual initiation, inadvertent initiation or manual startup for the HPCS system to start and inject into the reactor vessel. Also, there is a high level interlock for the HPCS injection valve to prevent overfilling the reactor vessel. This high level interlock is not normally overridden (this is done only during special evolutions with appropriate controls in place). Even if the HPCS system is inadvertently started it should not overfill and pressurize the reactor due to the high level interlock.

The Standby Liquid Control (SBLC) is another high pressure system used to shut down the reactor if the control rods fail to insert. The SBLC system has no auto start function so it is unlikely that a spurious initiation could occur. The SBLC system must be manually initiated by a key lock switch. The shift manager maintains custody of the keys. SBLC is a low flow rate system (about 42 gpm per pump) and is limited to the amount of water that is contained in the storage tank (about 5000 gallons). Even if the SBLC system was manually initiated and not monitored there would not be enough water in the storage tank to fill the reactor from normal water level and would not, therefore, pressurize the reactor.

The Reactor Feed pumps are the high pressure makeup system during normal operation. The Reactor Feed pumps are steam driven and cannot be operated when the reactor is in cold shutdown because no steam is available to drive the turbine. The Reactor Feed pumps also have a reactor hi level trip.

The Condensate system is the supply source to the Reactor Feed pumps. The Condensate pumps have a discharge pressure of about 150 psig and the Condensate Booster pumps have a discharge pressure of about 650 psig. During operation of both Condensate and Condensate Booster pumps, sufficient temperature margin is provided to ensure that the Technical Specification for the reactor pressure-temperature is not exceeded. This is accomplished by plant procedures dictating when Condensate and Condensate Booster operation is allowed. When the plant is in cold shutdown, reactor temperature is maintained above 70°F per Technical Specifications. If a Condensate pump was started (requires manual action) and lined up for injection and the resulting reactor level increase ignored, the reactor pressure-temperature limit would still not be exceeded since the shut off head of the Condensate pump is about 280 psig (about 300 psig required to exceed limit).

For the reactor pressure-temperature limit to be exceeded, a Condensate pump would have to be manually started, a Condensate Booster pump would have to be manually started, and both manually lined up for injection. Then the injection would have to be ignored by the operating crew and allowed to continue until the reactor is then pressurized above the pressure-temperature limits. The operating crew would have numerous indications that Condensate was injecting (feedflow indicators and recorders, check valve indication) and reactor level and pressure increases (Upset and shutdown level indication and recorders, narrow and wide range pressure indicators and recorders). Because of the number of operator errors that would have to occur and the number of indications that would have to be ignored, the probability of this event is very low.

The Low Pressure Core Spray (LPCS) system is a low pressure ECCS spray system. Technical Specifications for the reactor pressure temperature limit permit pressures from about 195 psig up to about 300 psig at temperatures from 70 up to 130°F. Above 130°F, pressures permitted by Technical Specifications increase immediately to above 575 psig and thereafter increase rapidly with temperature increases. The LPCS system has a discharge pressure of about 500 psig. Plant procedures also specify that temperatures normally be maintained between 120°F and 130°F during shutdown (temperatures are allowed to range from 70°F to 200°F in mode 4 and from 70°F to 140°F in mode 5 with approval of the shift manager).

During refueling outages there are typically two time periods when coolant temperature could be less than 100°F: (1) approximately 24 hours at the beginning of the outage when the vessel head is being detensioned; and (2) approximately 48 hours toward the end of the outage when the vessel head is being retensioned. As soon as the vessel head is retensioned, the IOIs instruct the operators to begin heatup for the reactor vessel inservice leak test.

In the event of an inadvertent LPCS actuation during times with the vessel head tensioned and coolant temperature below 100°F, LPCS and reactor vessel level instrumentation and alarms are available to provide operators with immediate information pertaining to system and reactor level conditions. Procedural controls are in place to prevent or mitigate a cold water overpressure event. Specifically, the refueling procedure requires one safety-relief valve (SRV) to be functional until the vessel head is removed. In case of a LPCS actuation, operators could open this SRV to prevent pressurizing the vessel. The procedure also contains a cautionary statement that directs operators upon an inadvertent LPCS actuation to immediately evaluate adequate core cooling and secure the LPCS pump. Therefore, operators would have to ignore procedure cautions, plant instrumentation, and alarms for a cold water overpressure event to occur from an inadvertent LPCS actuation.

Procedural controls and the short period of time when the vessel coolant temperatures could be below 100°F along with the number of indications that would have to be ignored make the probability of a cold water overpressure event due to an inadvertent actuation of this system very low.

The Low Pressure Core Injection (LPCI) systems (3 total) are low pressure ECCS injection systems. The LPCI systems have a discharge pressure of about 300 psig. If they were to be inadvertently initiated or manually started and lined up to the reactor they would only pressurize the reactor to approximately 300 psig. Technical Specifications requires that reactor metal temperature be maintained above 70°F. Because of this, the Technical Specification requirement for reactor pressure-temperature limit would not be exceeded. The procedural controls, plant equipment, instrumentation, and alarms pertaining to an inadvertent actuation of LPCS, as discussed above, are also in place to prevent a cold water overpressure event in case of an inadvertent LPCI actuation.

The Control Rod Drive (CRD) system is a high pressure system used to operate control rods. The CRD system is a low flow rate system with about 60 gpm flow rate to the reactor. During cold shutdown conditions reactor water level is maintained with CRD (makeup) and Reactor Water Cleanup (reject). Per plant procedures the reactor head vents are open when reactor coolant temperature is less than 190°F. During cold shutdown conditions the operators closely monitor reactor water level, temperature, and pressure. With the CRD flowrate low and the reactor head vents open, the operators should have sufficient time to react to regain control of reactor pressure, should any abnormalities occur.

Post Outage Primary System Hydrostatic Testing is another postulated over-pressurization event. GGNS has plant procedures as well as Technical Specifications that dictate parameters and steps in performing hydrostatic testing. Hydrostatic testing is considered an "Infrequently Performed Test or Evolution". This requires management oversight, crew briefs, review of industry events and assigned responsibilities for the test to be performed. Reactor coolant is heated up to 155 -175°F before reactor pressure is increased to test pressure. Reactor level is maintained with CRD (make-up) and/or RWCU (blowdown). Reactor pressure changes are limited to 50 psi per minute by plant procedures. Two safety relief valves are required to be operable during the test by plant procedures. Because of these strict controls, the likelihood of an overpressurization event during a hydrostatic test is minimal.

<u>Procedural Controls and Operator Training That Prevent Reactor Pressure Vessel Cold Over-Pressurization:</u>

Plant procedures and Technical Specifications dictate bands at which reactor water level, pressure and temperature are to be maintained which ensures an adequate level of safety during all modes of operation. Operation of GGNS follows the steam saturation curve. Therefore, the operating temperatures are expected to be well in excess of the minimum temperatures required by Technical Specifications. The control room operators are required by procedure to maintain reactor parameters (i.e., water level, pressure and temperature) within these bands and to frequently monitor those parameters. They are also required by procedure to report to the control room supervisor anytime operation is outside of a prescribed band. The control room supervisor is responsible to ensure that actions are taken to establish those parameters back within the desired band. Also, as previously noted, plant procedures require pre-job briefings and contingency plans before infrequent tests or evolutions are performed. Training reinforces these requirements in both classroom and simulator training. Finally, plant conditions, status of plant equipment, special activities along with their potential effect on key plant parameters, and contingency planning are discussed with oncoming crews during shift turnover.

At GGNS, work performed during an outage is scheduled by the Outage Management group. Outage Management includes Senior Reactor Operators who provide oversight of the outage schedule development to avoid conditions that could adversely affect reactor water level, pressure or temperature. From the outage schedule, a plan of the day is developed listing the work activities that will be performed that day. The plan of the day schedule is approved and reviewed by management. The plan of the day is assessed for shutdown risk to ensure an adequate level of safety is maintained. Any changes to the plan of the day must be approved by management.

The Refueling Integrated Operating Instruction (IOI) procedure requires that the reactor be depressurized before flooding up to the cold shutdown water level of about 230 inches when entering Refuel operations. Shutdown IOI requires that the Reactor Head Vents be opened when reactor coolant temperature is about 190°F during reactor cooldown. During Hydrostatic testing, the Reactor Vessel In-Service Leak Test IOI requires reactor coolant temperature be heated up to 155-175°F and at least 2 Safety Relief Valves to be operable prior to increasing reactor pressure. All of these help ensure the Technical Specifications requirements for reactor pressure-temperature limits are not exceeded.

IV. PROPOSED ALTERNATIVE

Examination Scope

As recommended by BWRVIP-05 and supported by the NRC Safety Evaluation dated July 28,1998, the proposed alternative eliminates the examinations required by 10 CFR 50.55a(g)(6)(ii)(A) and ASME Section XI, IWB-2500, for Examination Category B-A, Item No. B1.11. The proposed alternative is requested for the remainder of the current GGNS operating license.

Examination of the longitudinal (axial) reactor pressure vessel shell welds (Examination Category B-A, Item No. B1.12) shall be performed for 100 % of the welds. Axial weld examination shall also include that portion of the circumferential weld that intersects each axial weld, or approximately 2 to 3% of the intersecting circumferential weld.

The procedures and personnel for these examinations will meet the requirements of ASME Section XI, Appendix VIII as required by 10 CFR 50.55a(g)(6)(ii)(c).

Successive Examination of Flaws

For flaws detected in the circumferential weld (Item No. B1.11), the successive examinations required by ASME Section XI, IWB-2420 are not required for non-threatening flaws (e.g., embedded flaws from material manufacturing or vessel fabrication which experience negligible or no growth during the design life of the vessel), provided the following conditions are met:

a) The flaw is characterized as subsurface in accordance with BWRVIP-05, Figure 9-1,

b) The NDE technique and evaluation that detected and characterized the flaw as originating from the material manufacture or vessel fabrication is documented in a flaw evaluation report,

c) The vessel weld containing the flaw is acceptable for continued service in accordance with IWB-3600 and the flaw is demonstrated acceptable for the intended service life of the vessel.

Successive examination of flaws detected in the axial welds (Item No. B1.12), shall be performed as required by ASME Section XI, IWB-2420.

Additional Examinations of Flaws

For flaws detected in the circumferential weld (Item No. B1.11), the additional examinations required by ASME Section XI, IWB-2430 are not required provided the following conditions are met:

1. If the detected flaw is characterized as subsurface, additional examinations are not required.

2. If the flaw is not characterized as subsurface, then an engineering evaluation shall be performed addressing the following (at 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.

If the flaw meets the criteria of IWB-3600 for the intended service life of the vessel, then additional examinations may be limited to those welds subject to the same 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 if no failure mechanism exists, then no additional examinations are required.

For flaws detected in axial welds (Item No. B1.12), additional examinations shall be performed as required by ASME Section XI, IWB-2430.

Examination of the circumferential welds shall be preformed if axial weld examinations reveal an active, mechanistic mode of degradation. The NRC shall approve the timing and scope of these examinations.

V. CONCLUSION

Based on BWRVIP-05, the BWRVIP's response to the NRC's RAI, the plant-specific analysis performed by the NRC staff, and the discussion contained within, the proposed alternative to the cited requirements under the provisions of 10 CFR 50.55a(a)(3)(i) to eliminate the described examinations for the remainder of the current license is reasonable and will provide an acceptable level of quality and safety.

Table 1

GGNS RPV Shell Weld Information

Variable	USNRC	Variable Value by Weld Seam Identification		
Weld Seam[1]/ Welding Process	**LIMITING PLANT SPECIFIC ANALYSIS PARAMETERS AT 32 EFPY SER TABLE 2.6-4**	**AB (Lower Circ. Seam) SAW Heat 4P7216**	**AC (Upper Circ. Seam) SAW Heat 5P6771**	**AC (Upper Circ. Seam) SMAW Heat 412L4711**
Fluence @ 32 EFPY	0.510 x 10^{19} n/cm^2	0.250 x 10^{19} n/cm^2	0.250 x 10^{19} n/cm^2	0.250 x 10^{19} n/cm^2
Initial RT_{NDT}	-65°F	- 40.0°F	- 20.0°F	- 60.0°F
Weld Chemistry Factor	109.5	41	54	27
Weld Copper Content	0.10 wt%	0.03 wt%	0.04 wt%	0.02 wt%
Weld Nickel Content	0.99 wt%	0.81 wt%	0.95 wt%	0.91 wt%
Increase in Reference Temperature due to Irradiation (ΔRT_{NDT})	109.5°F	25.58°F	33.69°F	16.84°F
Margin Term	Not provided in Table 2.6-4	25.58°F	33.69°F	16.84°F
Mean Adjusted Reference Temperature. (Mean ART)	44.5°F	- 14.42°F	13.69°F	- 43.16°F
Upper Bound Adjusted Reference Temperature (Upper Bound ART)	Not provided in Table 2.6-4	11.16°F	27.38°F	-26.32°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.