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CNRO-2005-00036

August 24, 2005

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, DC 20555-0001

SUBJECT: River Bend Station
Docket No. 50-458
License No. NPF-47
Request for Alternative to 10 CFR 50.55a Examination Requirements of Category B1.11 Reactor Pressure Vessel Welds

REFERENCES:

1. 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*
2. NRC Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925), dated July 28, 1998
3. NRC Letter to Entergy Operations, Inc., dated April 11, 2001

Ladies and Gentlemen:

In November 1998, the NRC issued Generic Letter 98-05 (Reference 1) to inform addressees that the NRC had completed its review of the Boiling Water Reactor Vessel Internals Project (BWRVIP) final report, BWRVIP-05 (Reference 2). The generic letter informed licensees that they may request permanent (i.e., for the remaining term of operation under the existing initial license) relief from the inservice inspection (ISI) 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 the license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in Reference 2, and

(2) Licensees have implemented operator training and established procedures that limit the frequency of cold over-pressurization events to the amount specified in Reference 2.

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In accordance with Generic Letter 98-05 and pursuant to 10 CFR 50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) proposes Request for Alternative RBS-ISI-004 for River Bend Station (RBS). RBS-ISI-004 (see attachment) provides the information requested by the NRC to demonstrate that their plant-specific analysis adequately bounds the conditions of RBS. As demonstrated by BWRVIP-05, its NRC safety evaluation, and the discussion provided in RBS-ISI-004, Entergy believes this request provides an acceptable level of quality and safety. Following approval of this request, circumferential welds AA, AB, AC, and AD of the reactor pressure vessel shell will not require the examinations currently specified by ASME Section XI for the remainder of the current operating license.

A similar request was approved by the NRC for Grand Gulf Nuclear Station in April, 2001 (TAC No. MA9787) (Reference 3).

Entergy requests NRC approval by mid-August of 2006 to enable preparation for RBS' fall of 2007 refueling outage. Should you have any questions regarding this submittal, please contact Bill Brice at (601) 368-5076.

No new commitments are made in this letter.

Very truly yours,

FGB/WBB/bal
Attachment: Request for Alternative RBS-ISI-004

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CNRO 2005-00036

ATTACHMENT

REQUEST FOR ALTERNATIVE

RBS-ISI-004

ENTERGY OPERATIONS, INC.
RIVER BEND STATION
REQUEST FOR ALTERNATIVE
RBS-ISI-004

I. COMPONENT / EXAMINATION IDENTIFICATION

Component / Number: Reactor / Q1B13D001

Code Class: 1

References:

1. ASME Section XI, 1992 Edition, IWB-2500
2. 10 CFR Sections 50.55a(a)(3)(i) and 50.55a(g)(6)(ii)(A)(2)
3. BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection"
4. BWRVIP Response to NRC Request for Additional Information (RAI) on BWRVIP-05, 12/22/97
5. 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"
6. NRC Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report to (TAC No. M93925), dated July 28, 1998
7. NRC letter to Entergy Operations, Inc., dated March 26, 1999 (TAC No. MA0621)

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

Unit/Inspection Interval: River Bend Station (RBS) / Second (2nd) 10-year Interval

II. CODE REQUIREMENTS

The ASME code of record for the second 10-year interval at RBS is the 1992 Edition with portions of the 1993 Addenda. ASME Section XI, 1992 Edition, IWB-2500 requires the reactor pressure vessel (RPV) circumferential shell 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, IWB-2420 requires the sequence of examinations established in the first interval to be repeated during subsequent intervals to the extent practical.

III. PROPOSED ALTERNATIVE

Pursuant to the provisions of 10 CFR 50.55a(a)(3)(i) and consistent with guidance provided in NRC Generic Letter 98-05 (Reference 5), Entergy Operations, Inc. (Entergy) requests NRC approval to use an alternative from the examination of RPV circumferential welds required by ASME Section XI, IWB-2500, Examination Category B-A, Item No. B1.11 as described below. As a result of this request, circumferential welds AA, AB, AC, and AD of the reactor pressure vessel shell will not require the examinations currently specified by ASME Section XI for the remainder of the current operating license.

A. Examination Scope

Examination of the longitudinal (axial) RPV 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 procedure and personnel employed for these examinations will meet the requirements of ASME Section XI, Appendix VIII as required by 10 CFR 50.55a(g)(6)(ii)(C).

B. Successive Examination of Flaws

For flaws detected in the circumferential weld (Item No. B1.11), the successive examinations required by 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:

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

(2) The Non-destructive Examination (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.

(3) 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 RPV.

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

C. 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) A determination of the root cause of the flaw,

   (b) An evaluation of any potential failure mechanisms,

   (c) An evaluation of service conditions which could cause subsequent failure,

   (d) 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 IWB-2430.

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

IV. BASIS FOR ALTERNATIVE

The basis for this request is documented in the report *BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations* (BWRVIP-05) (reference 3) that was transmitted to the NRC in September 1995 and BWRVIP Response to NRC RAI on BWRVIP-05 (reference 4) that was transmitted to the NRC on December 18, 1997. The NRC staff approved BWRVIP-05 in a letter dated July 28, 1998 (Reference 6).

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 estimated to be end-of-life (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 this alternative, the following information is provided to show the conservatisms of the NRC plant-specific analysis as they apply to RBS:

- The region of the vessel that corresponds to the top and bottom of active fuel is designated as the beltline. The RBS beltline does not contain circumferential welds. The closest two weld seams are AB, which is approximately 7 inches below the beltline, and AC, which is approximately 19 inches above the beltline region. These two seams are the closest circumferential weld seams to the active fuel core and are, therefore, considered the limiting seams for this evaluation.

  The calculated 32 effective full power years (EFPY) EOL peak fluence (E>1.0 MeV) is 4.98E+18 n/cm^2. This peak is located at approximately 13 inches above the mid-plane of the vessel beltline region at the clad/base metal interface. The peak value decreases about 80% to 9.96E+17 n/cm^2 at the top and bottom of the beltline region. For purposes of this evaluation, conservative mean fluence value has been calculated for the two limiting seams by averaging the peak fluence with the top and bottom beltline fluence. This mean fluence value is 2.988E+18 n/cm^2. It is used to determine the Delta Reference Temperature for nil-ductility temperature (ΔRT_{NDT}) values and subsequently the mean RT_{NDT} values for each weld heat in the two circumferential weld Seams AB and AC. This value is conservative because beltline region fluences are used for non-beltline region materials.

- 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 (see Table 2.6-4 of Reference 6). The maximum (peak) inner surface fluence used for the RPV limiting <u>non-beltline</u> region at the EOL (32 EFPY) is predicted to be 0.2988E+19 n/ cm^2. Thus, the effect of fluence on embrittlement is lower, and the NRC analysis as described in the NRC safety evaluation (Reference 6) is conservative for RBS.

- Review of the RPV fabrication records also reveals that Heat No. 76916 (SMAW 5/32" E8018-G) was used only for weld pick-ups on Seam AB and is, therefore, considered to be a very small contributor to the overall weld metal properties for Seam AB. However, as shown in Table 1, the attributes of this specific weld heat causes Seam AB to be the limiting circumferential weld.

As described above, there is conservatism in the already low circumferential-weld-failure probabilities as related to RBS. Other 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, the limiting circumferential weld in the RBS RPV is Seam AB with a maximum calculated shift in RT_{NDT} (i.e., ΔRT_{NDT}) of 90°F and a mean reference temperature (i.e., mean RT_{NDT}) of 30°F at 32 EFPY (EOL). The values used by the NRC in the plant-specific analysis for ΔRT_{NDT} is 109.5°F with a resulting mean RT_{NDT} of 44.5°F. Therefore, it is evident that the values used by the NRC in their plant-specific analysis are bounding and provide additional assurance that the RPV welds are also bounded by the BWRVIP-05 report.

An added margin is also provided at RBS by the NDE of the RPV welds. The examination coverage for all welds, except for circumferential weld AA, exceeded 90% coverage of the full volume during the first 10-year examination. Weld AA examination coverage was limited to 62% weld volume. RPV geometric limitations created by the change in vessel diameter at the transition to the bottom head prevented greater than 90% weld volume coverage. This limitation was evaluated and approved by the NRC staff in a letter to RBS dated March 26, 1999 (TAC No. MA0621) (Reference 7).

In previous evaluations, the NRC concluded that beyond design basis events occurring during plant shutdown could lead to cold over-pressure events that could challenge RPV 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.

Entergy has reviewed the BWRVIP's response and concurs that the conditions and events are accurately depicted and that the procedures and personnel training at RBS 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 (Reference 6) and is conservative for RBS.

<u>Review of Potential Injection Sources That Could Cause an RPV Cold Over-Pressurization Event:</u>

The Reactor Core Isolation Cooling (RCIC) system is one of the high pressure make-up systems at RBS. The RCIC system utilizes a steam turbine-driven pump. RCIC injection during cold shutdown is not possible since no steam is available to drive the RCIC turbine.

The High Pressure Core Spray (HPCS) system is another high pressure make-up system at RBS. The HPCS pump is motor operated, so it can be operated when the reactor is in cold shutdown. However, the HPCS system would require manual initiation, inadvertent initiation, or manual startup to start and inject into the RPV. Also, there is a high RPV water level interlock for the HPCS injection valve to prevent overfilling the RPV. This high level interlock is not normally overridden (this is done only for valve stroke testing with the HPCS pump breaker racked out). Even if the HPCS system is inadvertently started, it would not overfill and pressurize the reactor due to the high level interlock.

The Standby Liquid Control System (SBLC) is a high pressure system used to shut down the reactor if the control rods fail to insert. SBLC has no auto start function so a spurious start is unlikely. SBLC must be manually initiated by a key lock switch. A Control Room Senior Reactor Operator (SRO) 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 was manually initiated and not monitored, there is not enough water in the storage tank to fill the RPV from normal water level and would not, therefore, pressurize it.

The Reactor Feed Pumps (RFP) provide high pressure make-up during normal operation. RBS has three 33% motor-driven pumps that are fed by the condensate pumps (CPs). These pumps provide full flow at pressures of about 1400 psig to the reactor. RBS has three 50% capacity CPs that supply the RFPs during normal operation. The CPs have a discharge pressure of about 550 psig. This corresponds to about 340 psig suction pressure at the RFP. The CPs have a shutoff discharge pressure of about 575 psig. During operation of the CPs, sufficient temperature margin is provided to ensure that the Technical Specification for the reactor pressure and temperature (P/T) is not exceeded. This is accomplished by plant procedures dictating when CP operation is allowed. When the plant is in cold shutdown, reactor temperature is maintained above 70°F per Technical Specifications.

For the reactor P/T limit to be exceeded, a CP would have to be manually started and 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 pressurized above the P/T limits. The operating crew would have numerous indications that condensate was injecting (feed flow indicators and recorders, check valve indication) and that reactor level and pressure were increasing (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) and Low Pressure Core Injection (LPCI) systems are low pressure emergency core cooling systems (ECCS). LPCI is actually a mode of the Residual Heat Removal system. Technical Specifications for the reactor P/T limit permit pressures from about 95 psig up to about 313 psig at temperatures from 70 up to approximately 120°F. Between 120°F and 150°F, the pressure permitted by Technical Specifications remains constant at about 313 psig. At 150°F, the allowable pressure increases immediately to roughly 540 psig and thereafter increases rapidly as temperature increases. LPCS has a discharge pressure of about 500 psig and LPCI has a discharge pressure of about 329 psig. During refueling outages, there is typically only a very short period of time during RPV head de-tensioning and following head re-tensioning in which an overpressurization event could occur. As soon as the head is re-tensioned, plant procedures instruct the operators to begin heat-up for the RPV inservice leak test. Temperatures are normally maintained between 90°F and 120°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 SRO). Therefore, the reactor bulk coolant temperature is normally well above 70°F. Procedural controls and the short period of time when the vessel coolant temperatures are low, make the probability for an overpressurization due to an inadvertent actuation of these systems very low. The automatic initiation signals for LPCI and LPCS are low reactor water level of -143" and high differential pressure between the drywell and containment of 1.68 psid. During the short periods around vessel head de-tensioning and re-tensioning, the reactor water level is maintained near the vessel flange which is 200" and the drywell head is removed making the Drywell and containment one area. Because of this, the possibility of an automatic initiation of these systems is very small.

The Control Rod Drive (CRD) system is a high pressure system used to operate the control rods. The CRD system is a low flow rate system with about 45 gpm flow rate to the reactor. During cold shutdown conditions, reactor water level is maintained with CRD (makeup) and the Reactor Water Cleanup System (RWCU) (reject). Per plant procedures, the RPV head vents are open when reactor coolant temperature is less than

190°F. During cold shutdown conditions, the operators closely monitor reactor water level, pressure, and temperature. With the CRD flowrate low and the RPV 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 a postulated over pressurization event. RBS has plant procedures as well as Technical Specifications that dictate parameters and steps when 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 >150°F before reactor pressure is increased to test pressure. Reactor level is maintained with CRD (make-up) and/or RWCU (reject). Reactor pressure changes are limited to 50 psi per minute by plant procedures. All safety relief valves are required to be operable during the test by plant procedures. Because of these strict controls, the likelihood of an over pressurization 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 that ensure an adequate level of safety during all modes of operation. Operation of RBS 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 SRO anytime operation is outside of a prescribed band. The SRO is responsible to ensure that actions are taken to return 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 RBS, work performed during an outage is scheduled by the Outage Management group. Outage Management includes SROs who provide oversight of the outage schedule development to avoid conditions that could adversely affect reactor water level, pressure, or temperature. The outage schedule is reviewed by the Outage Risk Assessment Team (ORAT) to assure a proper level of plant safety is maintained. Significant emergent work or significant changes to the schedule are also reviewed by the ORAT. The outage is performed following this schedule.

General Operating Procedure (GOP), "Power Decrease/Plant Shutdown", requires that the RPV head vents be opened when reactor coolant temperature is about 190°F during reactor cooldown and administrative control be established to ensure that the RPV remains vented. During hydrostatic testing, the Reactor Vessel In-Service Leak Test procedure requires reactor coolant temperature be heated up to >150°F and all Safety Relief Valves to be operable prior to increasing reactor pressure. These help ensure the Technical Specifications requirements for reactor P/T limits are not exceeded.

V. <u>CONCLUSION</u>

Based on BWRVIP-05, the BWRVIP's response to the NRC's Request for Additional Information, the plant-specific analysis performed by the NRC staff, and the discussions above, Entergy believes the proposed alternative to eliminate the described examinations is reasonable and provides an acceptable level of quality and safety. Therefore, Entergy requests the NRC staff authorize the use of RBS-ISI-004 in accordance with 10 CFR 50.55a(a)(3)(i).

This proposed alternative is for the remaining term of operation under the existing license.

Table 1

SUMMARY OF RESULTS

(see supplemental information)

Summary of Results for MEAN Reference Temperature (Mean RT_{NDT}) Evaluation for the River Bend Reactor Vessel Non-Beltline Circumferential AB and AC Seam Weld Materials by Heat Number at the Vessel I.D. Surface Applicable to 32 EFPY with Power Uprates.

Material Description		Chemical Composition				Inside Surface @ Vessel I.D. (Clad/Base Metal Interface)		
Vessel Circ. Weld Seam	Weld Material Identification	Cu wt%	Ni wt%	Initial RT_{NDT}	Chemistry Factor (CF)	32 EFPY (EOL) Fluence, n/cm^2 (E>1.0 Mev.)	ΔRT_{NDT}, F at 32 EFPY	Mean RT_{NDT}, F at 32 EFPY
River Bend Results								
AB	Raco/NMM (Tandem Wire) 4P7216 / Linde 124 / 0751	0.038	0.820	-80	51.4	2.988E+18	34.4	-45.6
AB	Raco/NMM (Single Wire) 4P7216 / Linde 124 / 0751	0.038	0.820	-50	51.4	2.988E+18	34.4	-15.6
AB	Raco/NMM (Tandem Wire) 4P7465 / Linde 124 / 0751	0.02	0.807	-60	27.0	2.988E+18	18.1	-41.9
AB	Raco/NMM (Single Wire) 4P7465 / Linde 124 / 0751	0.02	0.807	-60	27.0	2.988E+18	18.1	-41.9
AB	(SMAW Process) 02R486 / J404B27AG	0.07	0.99	-70	95.0	2.988E+18	63.6	-6.4
AB	(SMAW Process) 03L048 / B525B27AF	0.09	0.96	-60	122.0	2.988E+18	81.6	21.6

Material Description		Chemical Composition				Inside Surface @ Vessel I.D. (Clad/Base Metal Interface)		
Vessel Circ. Weld Seam	Weld Material Identification	Cu wt%	Ni wt%	Initial RT_{NDT}	Chemistry Factor (CF)	32 EFPY (EOL) Fluence, n/cm^2 (E>1.0 Mev.)	ΔRT_{NDT}, F at 32 EFPY	Mean RT_{NDT}, F at 32 EFPY
AB	(SMAW Process) 76916 / D516B27AE	0.10	0.95	-60	134.5	2.988E+18	[90.0]	[30.0]
AB	(SMAW Process) L83978 / J414B27AD	0.02	1.06	-80	27.0	2.988E+18	18.1	-61.9
AC	Raco/NMM (Tandem Wire) 5P6771 / Linde 124 / 0342	0.034	0.934	-20	46.2	2.988E+18	30.9	10.9
AC	Raco/NMM (Single Wire) 5P6771 / Linde 124 / 0342	0.034	0.934	-30	46.2	2.988E+18	30.9	0.9
AC	(SMAW Process) 640892 / J424B27AE	0.09	1.00	-60	122.0	2.988E+18	81.6	21.6
AC	(SMAW Process) 629865 / A421A27AD	0.05	1.10	-70	68.0	2.988E+18	45.5	-24.5
NRC Bounding Criteria for CB&I Group Results								
Circ. Weld	NRC Limiting Plant-Specific Weld Analysis Parameters at 32 EFPY SER Table 2.6-4, CB&I Group	0.10	0.99	-65	134.9	5.10E+18	109.5	44.5

[ ] Limiting/Controlling value of the Shift in the Reference Temperatures (ΔRT_{NDT}) and the Mean Reference Temperatures (Mean RT_{NDT}).

SUPPLEMENTAL INFORMATION RELATED to Table 1

1) RBS RPV beltline does not contain circumferential welds. Weld Seam AB is approximately 7 inches below the vessel beltline and Weld Seam AC is approximately 19 inches above the vessel beltline region. Weld Seams AB and AC are the two circumferential welds closest to the active fuel core and are therefore considered to be the limiting vessel girth welds in this evaluation.

2) The calculated 32 EFPY peak fluence value at the inner surface location is 4.98E+18 n/cm^2 located at 13 inches above the mid-plane of the core and corresponding vessel beltline region. This peak fluence value decreases to 9.96E+17 n/cm^2 (peak reduced by 80%) at the top and bottom locations of the vessel beltline region.

   Although the maximum inner surface fluence applicable outside of the vessel beltline region is 9.96E+17 n/cm^2 or lower, a mean fluence value of 2.988E+18 n/cm^2 for the vessel beltline is established using an average of the peak and reduced values for the beltline region.

   This alternative evaluation conservatively uses the 32 EFPY mean fluence value (2.988E+18 n/cm^2) that applies to the vessel beltline region to determine the ΔRT_{NDT} values and subsequently the Mean RT_{NDT} values for each weld heat within the two circumferential seams AB and AC, which are non-beltline materials. In addition, no credit is taken for the attenuation caused by the RPV inner surface cladding.

3) Initial RT_{NDT} determined in accordance with ASME Section III, NB-2300.

4) The largest 32 EFPY ΔRT_{NDT} and Largest 32 EFPY mean RT_{NDT} occurs in the SMAW portion (Ht No. 76916) of weld AB; therefore, this weld is the limiting of the two circumferential welds and is used to compare against the NRC's plant-specific analysis 32 EFPY parameters for vessels fabricated by CB&I. This Request for Alternative has compared the shift of Weld Seam AB to the shift predicted by the NRC for the CB& I Fabrication Group, and the RBS shift is less. As such, the NRC analysis is bounding and provides basis for this alternative.

5) Mean RT_{NDT} is determined using Eq. 1 of Reg. Guide 1.99 excluding the Margin Term in Eq. 4. ΔRT_{NDT} is determined using Eq. 2. Eq. 3 is not used because the evaluation is for the inner surface of the vessel (no attenuation into the vessel).

6) Based on BWRVIP-135, there are no surveillance data sets for any of the weld heats identified in Table 1. However, the best estimate chemistry values provided in Table D-1 of BWRVIP-135 were used to determine the Chemistry Factors used in Table 1 for weld heats 4P7216, 4P7465, and 5P6771.