July 14, 2006

Mr. Paul D. Hinnenkamp
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
5485 US Highway 61N
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - RELIEF REQUEST RBS-ISI-004 FOR ALTERNATIVE TO SECTION 50.55a OF TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS* (10 CFR) FOR EXAMINATION REQUIREMENTS OF CATEGORY B1.11 REACTOR PRESSURE VESSEL WELDS (TAC NO. MC8201)

Dear Mr. Hinnenkamp:

By letter dated August 24, 2005, as supplemented by letter dated February 8, 2006, Entergy Inc. (Entergy, or the licensee), submitted Relief Request RBS-ISI-004 requesting relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection [(ISI)] of Nuclear Power Plant Components," requirements related to examination of reactor pressure vessel (RPV) circumferential shell welds at River Bend Station (RBS). Entergy requested relief from the augmented ISI requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g)(6)(ii)(A)(2) for the volumetric examination of circumferential RPV welds (ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.11, Circumferential Shell Welds). The licensee proposed an alternative pursuant to 10 CFR 50.55a(a)(3)(i), to the RPV circumferential shell welds examination requirements of the ASME Code, Section XI, for the remaining term of facility operating license NPF-47 for RBS.

The NRC staff has reviewed your request, and, based on the information provided, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety for the remaining term of facility operating license NPF-47 for RBS. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i).

Additional requirements of the ASME Code, Section XI, for which relief has not been specifically requested remain applicable, including third party reviews by the Authorized Nuclear Inservice Inspector.

The NRC staff's detailed technical review and conclusions are documented in the enclosed safety evaluation.

If you have any questions related to this issue, please contact me at 301-415-3308.

Sincerely,

/RA/

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE FOR EXAMINATION REQUIREMENTS OF

CATEGORY B1.11 REACTOR VESSEL CIRCUMFERENTIAL WELDS

RELIEF REQUEST RBS-ISI-004

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated August 24, 2005, as supplemented by letter dated February 8, 2006, Entergy Inc. (Entergy, or the licensee), submitted Relief Request RBS-ISI-004 requesting relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection [(ISI)] of Nuclear Power Plant Components," requirements related to examination of reactor pressure vessel (RPV) circumferential shell welds at River Bend Station (RBS). Entergy requested relief from the augmented ISI requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g)(6)(ii)(A)(2) for the volumetric examination of circumferential RPV welds (ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.11, Circumferential Shell Welds) for the remaining term of facility operating license NPF-47 for RBS. Specifically, the request is regarding the circumferential welds AA, AB, AC, and AD of the RPV requiring the examinations currently specified by ASME Code, Section XI.

The relief request was regarding the use of a proposed alternative pursuant to the provisions of 10 CFR 50.55a(a)(3)(i), in accordance with Boiling Water Reactor (BWR) Vessel and Internals Project (BWRVIP)-05, “BWR Vessel and Internals Project, BWR RPV Shell Weld Inspection Recommendations,” TR-105697, September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, May 16, June 4, June 13, and December 18, 1997, and January 13, 1998 (Reference 1), the U.S. Nuclear Regulatory Commission (NRC) staff’s Final Safety Evaluation Report of the Boiling Water Reactor Vessel and Internals Project (BWRVIP)-05 report dated July 28, 1998 (Reference 2, hereafter called “BWRVIP-05 SER”), and NRC Generic Letter (GL) 98-05, “Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examinations Requirements on Reactor Pressure Vessel Circumferential Shell Welds,” November 10, 1998 (Reference 3), to the RPV circumferential shell welds examination requirements of the ASME Code, Section XI for the remaining term of facility operating license NPF-47 for RBS.

2.0 REGULATORY EVALUATION

The ISI of the ASME Code Class 1, 2, and 3 components is performed in accordance with the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," and applicable Addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Paragraph 50.55a(a)(3) of 10 CFR states that proposed alternatives to the requirements of paragraph (g) may be used, when authorized by the staff, if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), components (including supports) which are classified as ASME Code Class 1, 2, and 3 shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulation in paragraph 10 CFR 50.55a(g)(4) require that inservice examination of components and system pressure tests conducted during the first 120-month interval and subsequent intervals comply with the requirements in the latest Edition and Addenda of ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

The regulation in paragraph 50.55a(g)(6)(ii)(A)(2) of 10 CFR requires that all licensees augment their RPV examination by implementing once, as part of the ISI interval in effect on September 8, 1992, the examination requirements for RPV shell welds specified in Item No. B1.10, Examination Category B-A, “Pressure Retaining Welds in Reactor Vessel,” of Table IWB-2500-1 to Section XI. Additionally, the regulation in paragraph 50.55a(g)(6)(ii)(A)(2) of 10 CFR requires that the examinations cover “essentially 100 percent” of the RPV shell welds.

The regulation in paragraph 10 CFR 50.55a(g)(6)(ii)(A)(2) defines an "essentially 100 percent” examination as covering 90 percent or more of the examination volume of each weld. The schedule for implementation of the augmented inspection is dependent upon the number of months remaining in the 10-year ISI interval that was in effect on September 8, 1992.

The applicable ISI Code of record for RBS is the 1992 Edition with portions of 1993 Addenda of ASME Code, Section XI, which was approved by the staff in a letter from W. Beckner to J. Dewease dated December 12, 1996.

2.1 Regulatory Background

2.1.1 BWRVIP-05 Report

By letter dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, May 16, June 4, June 13, and December 18, 1997, and January 13, 1998, the BWRVIP, a technical committee of the BWR Owners Group (BWROG), submitted the BWRVIP-05 report (Reference 1) for NRC staff review. This report evaluated the current inspection requirements for RPV shell welds in BWRs, formulated recommendations for alternative inspection requirements, and provided a technical basis for these recommended requirements. As modified, the BWRVIP-05 report proposed to reduce the scope of inspection of BWR RPV welds from essentially 100 percent of all RPV shell welds to examination of 100 percent of the axial (i.e., longitudinal) welds and essentially 0 percent of the circumferential RPV shell welds, except for the intersections of the axial and circumferential shell welds. In addition, the report included proposals to provide alternatives to the ASME Code, Section XI requirements for successive and additional examinations of circumferential shell welds, provided in paragraphs IWB-2420 and IWB-2430, respectively, of the ASME Code, Section XI.

On July 28, 1998, the staff issued a Safety Evaluation Report (SER) on the BWRVIP-05 report[1] (Reference 2). This evaluation concluded that the failure frequency of RPV circumferential shell welds in BWRs was sufficiently low to justify elimination of ISI of these welds. In addition, the evaluation concluded that the BWRVIP proposals on successive and additional examinations of circumferential shell welds were acceptable. The evaluation indicated that examination of the circumferential shell welds shall be performed if axial shell weld examinations reveal an active degradation mechanism.

In the BWRVIP-05 report, the BWRVIP committee concluded that the conditional probabilities of failure for BWR RPV circumferential shell welds are orders of magnitude lower than that of the axial shell welds. As a part of its review of the report, the NRC staff conducted an independent probabilistic fracture mechanics assessment of the results presented in the BWRVIP-05 report. The NRC staff's assessment conservatively calculated the conditional probability of failure from RPV axial and circumferential shell welds during the initial (current) 40-year license period and at conditions approximating an 80-year RPV lifetime for a BWR nuclear plant, as indicated respectively in Tables 2.6-4 and 2.6-5 of the NRC staff's July 28, 1998 SER. The failure frequency for an RPV is calculated as the product of the frequency for the critical (limiting) transient event and the conditional probability of failure for the weld.

The NRC staff determined the conditional probability of failure for axial and circumferential shell welds in BWR RPVs fabricated by Chicago Bridge and Iron (CB&I), Combustion Engineering (CE), and Babcock and Wilcox (B&W). The analysis identified a low temperature overpressurization (LTOP) event that occurred in a foreign reactor as the limiting event for BWR RPVs, with the pressure and temperature from this event used in the probabilistic fracture mechanics calculations. The NRC staff estimated that the probability for the occurrence of the LTOP transient was 1×10^{-3} per reactor-year. For each of the RPV fabricators, Table 2.6-4 of the BWRVIP-05 SER identified the conditional failure probabilities for the plant-specific conditions with the highest projected reference temperature (for that fabricator) after the initial 40-year license period.

2.1.2 Generic Letter (GL) 98-05

The NRC GL 98-05 states that BWR licensees may request permanent (i.e., for the remaining term of operation under the existing, initial license) relief from the ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential RPV shell welds (ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, "Circumferential Shell Welds") by demonstrating the compliance with the two criteria described below:

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[1] The NRC staff has identified that in some instances the Staff SER is referenced as dated on July 28, 1998, others in July 30, 1998. For clarification purposes the NRC staff notes that this SER is a letter addressed to Carl Terry, BWRVIP Chairman, dated July 28, 1998 and titled "Final SER of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925)."

(1) At the expiration of the license, the circumferential shell welds will continue to satisfy the limiting conditional failure probability for circumferential shell welds in the BWRVIP-05 SER, and

(2) Licensees have implemented operator training and established procedures that limit the frequency of LTOP events to the amount specified in the BWRVIP-05 SER.

Additionally, GL 98-05 states that the licensees will still need to perform the required inspections of "essentially 100 percent" of all axial shell welds.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Components/Examinations Affected (As Submitted)

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 Catagory: B-A

Item No.: B1.11

Examination Required: Volumetric Examination of Welds and Adjacent Base Materials

Description: Circumferential Shell Welds in Reactor Pressure Vessel [Welds AA, AB, AC, and AD]

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

3.2 Applicable Code and Regulatory Requirements for which Relief is Requested

The ASME Code of record for the second 10-year interval at RBS is the 1992 Edition with portions of the 1993 Addenda. The licensee requested relief from the requirements of the ASME Code, Section XI, 1992 Edition through portions of 1993 Addenda, Subarticle IWB-2500, Table IWB 2500-1, Examination Category B-A, “Pressure Retaining Welds in Reactor Vessel.”

In addition to the ASME Code, Section XI volumetric examinations, licensees are required to perform a one-time volumetric examination (augmented examination) of the RPV shell welds as required by 10 CFR 50.55a(g)(6)(ii)(A). The licensee’s Relief Request RBS-ISI-004 has requested relief from the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A) for the RPV circumferential shell welds.

3.3 Licensee’s Proposed Alternative Examination (As Submitted)

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[2]), Entergy Operations, Inc. (Entergy) requests NRC approval to use an alternative from the examination of RPV circumferential welds required by ASME [Code,] 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 RPV shell will not require the examinations currently specified by ASME [Code,] 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 [Code,] 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

[2] The reference numbers in the “As submitted” sections of this safety evaluation (SE) refer to the references in Section 3.1 of this SE.

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 [Code,] 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.

3.4 Licensee's Basis for Relief (As Submitted)

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 [request for additional information] 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 [an] order[s] of magnitude lower than that of the axial shell welds. Additionally, the NRC safety evaluation [SE] 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 decreases slightly more at the bottom of the beltline region. For purposes of this evaluation, a conservative mean fluence value has been calculated for the two limiting seams by averaging the peak fluence with the top of 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.

The fluence was determined based on the vessel fluence analysis results reported in "Neutron Transport Analysis for River Bend Station" (MPM-904779). This analysis was performed by MPM Technologies using methods consistent with the requirements of Regulatory Guide 1.190. MPM methods were previously described in "Benchmarking of Nine Mile Point Unit 1 and Unit 2 Neutron Transport Calculations" (MPM-402781 revision 1) and approved in "Nine Mile Point Nuclear Station, Unit 1 - Issuance of Amendment RE: Pressure-Temperature Limit Curves and Tables", (TAC NO. MB6687). Additionally, a[n] RBS specific application of these methods is described in "BWR Vessel and Internals Project River Bend 183 Degree Surveillance Capsule Report" (BWRVIP-113) which was provided to the NRC by letter of June 10, 2003. The River Bend fluence calculations were updated in November, 2004 (MPM-904779) to include shroud fluence data to support scheduled inspections as described below. Both RBS fluence analyses (BWRVIP-113 and MPM-904779) were performed by MPM Technologies and use the same transport methods. The more recent calculation (MPM-904779) used more detailed source term data. The data was revised in the following four areas:

1) MPM-904779 includes eight separate transport calculations spread throughout the cycle 1 core's operation to support the cycle 1 dosimetry evaluation. BWRVIP-113 uses a single cycle average calculation since the initial cycle dosimetry was not evaluated.

2) Both MPM-904779 and BWRVIP-113 evaluate each cycle using cycle average input to the transport calculations except for cycles just prior to dosimetry removal. The BWRVIP-113 calculation used the 3D moderator density distribution from the middle of cycle point with the power distribution closest to the cycle average distribution. MPM-904779 calculated a weighted average moderator density distribution from 3D moderator distributions calculated throughout core life.

3) MPM-904779 included actual operational data for cycles 1 - 12 while BWRVIP- 113 only used operational data for cycles 1 - 9. MPM-904779 used a slightly more conservative fuel cycle to extrapolate the fluence to end of plant life.

4) BWRVIP-113 was performed in 1/8 core symmetry while MPM-904779 used ¼ core symmetry. This change more correctly models the jet

pumps which are ¼ core symmetric. The 183 degree capsule dosimetry is not impacted since the asymmetry occurs approximately 90 degrees from the capsule location. At the position of the highest fluence the asymmetry is less than 4%.

The refinements included in MPM-904779 result in an increase in accuracy as illustrated by Table 1 and a slight increase in the maximum vessel fluence. The 32 EFPY maximum inside radius fluence increased from 4.38E+18 to 4.98E+18. This difference is less than the uncertainties of either calculation and, therefore, the methods used in the reports are considered equivalent. However, the MPM-904779 results are preferred based on the improved accuracy.

Table 1: Dosimetry Comparisons

	Calculated/Measured Ratio for River Bend	
	BWRVIP-113	MPM-904779
Cycle 1 Flux Wire		
Fe Total	N/A	1.04
Cycle 9		
Cu Total	0.803	0.872
Fe Total	0.914	1
Cycle 9 Capsule Total	0.858	0.936

- 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 non-beltline 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 2, 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 2.

As shown in Table 2, 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 EF and a mean reference temperature (i.e., mean RT_{NDT}) of 30 EF at 32 EFPY (EOL). The values used by the NRC in the plant-specific analysis for ΔRT_{NDT} is 109.5 EF with a resulting mean RT_{NDT} of 44.5 EF. 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 gallons per minute 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 [pounds per square inch gauge] 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 [a] cold shutdown, reactor temperature is maintained above 70 EF 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 EF up to approximately 120 EF. Between 120 EF and 1500 EF, the pressure permitted by [the] Technical Specifications remains constant at about 313 psig. At 1500 EF, 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 retensioning in which an overpressurization event could occur. As soon as the head is retensioned, plant procedures instruct the operators to begin heat-up for the RPV inservice leak test. Temperatures are normally maintained between 90 EF and 120 EF during shutdown (temperatures are allowed to range from 70 EF to 200 EF in Mode 4 and from 70 EF to 140 EF in Mode 5 with approval of the SRO). Therefore, the reactor bulk coolant temperature is normally well above 70 EF. 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 [a] low reactor water level of -143" and [a] high differential pressure between the drywell and containment of 1.68 psid [pounds per square inch differential]. 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 EF. 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 EF 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 [ponds per square inch] 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 EF 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 EF and all Safety Relief Valves to be operable prior to increasing reactor pressure. These [this] help ensure [s] the Technical Specifications requirements for reactor P/T limits are not exceeded.

Table 2
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		Initial RT_{NDT}	Chemistry Factor (CF)	Inside Surface @ Vessel I.D. (Clad/Base Metal Interface)		
Vessel Circ. Weld Seam	Weld Material Identification	Cu wt %	Ni wt %			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

Material Description		Chemical Composition		Initial RT_{NDT}	Chemistry Factor (CF)	Inside Surface @ Vessel I.D. (Clad/Base Metal Interface)		
Vessel Circ. Weld Seam	**Weld Material Identification**	**Cu wt %**	**Ni wt %**			**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	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
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

Material Description		Chemical Composition		Initial RT_{NDT}	Chemistry Factor (CF)	Inside Surface @ Vessel I.D. (Clad/Base Metal Interface)		
Vessel Circ. Weld Seam	**Weld Material Identification**	**Cu wt %**	**Ni wt %**			**32 EFPY (EOL) Fluence, n/cm^2 (E>1.0 Mev.)**	**ΔRT_{NDT}, F at 32 EFPY**	**Mean RT_{NDT}, F at 32 EFPY**
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 2

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 [Code,] 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 [by] 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 2. However, the best estimate chemistry values provided in Table D-1 of BWRVIP-135 were used to determine the Chemistry Factors used in Table 2 for weld heats 4P7216, 4P7465, and 5P6771.

3.5 Evaluation

As described previously, GL 98-05 provides two criteria that must be satisfied when BWR licensees request relief from ISI requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential RPV welds (ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, Circumferential Shell Welds). These criteria are intended to demonstrate that the conditions at the licensee's plant are bounded by those in the BWRVIP-05 SER. The licensee will still need to perform the required inspections of "essentially 100 percent" of all axial shell welds.

3.5.1 Criteria 1: Circumferential Weld Conditional Failure Probability

The NRC staff evaluated the licensee's conformance with Criterion 1 of GL 98-05 using the guidance provided in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001 (Reference 4).

3.5.1.1 Review of Neutron Fluence Calculations

The NRC staff provided guidance in RG 1.190 for calculating EOL neutron fluence in an acceptable manner. The NRC staff's review of the methodology of fluence calculation was conducted in the following steps:

(1) Review of the Methodology of the Neutron Fluence Calculation for Acceptability:

The licensee used a fluence calculation methodology previously approved by NRC on a plant-specific basis for "Nine Mile Point Nuclear Station, Unit 1 - Issuance of Amendment Re: Pressure-Temperature Limit Curves and Tables (TAC No. MB6687)." MPM Technologies submitted a report entitled, "Benchmarking of Nine Mile Point Unit 1 and Unit 2 Neutron Transport Calculations (MPM-402781)" (Reference 5), to the NRC staff. The methods that are used in this report rely on S_8 angular quadrature and P_3 legendre polynomial expansion. The licensee used the BUGLE-96 cross-section library. Additionally, the licensee's methodology included the synthesis of two, two-dimensional calculations to form a three-dimensional fluence model. As established in RG 1.190, these approaches comprise an acceptable fluence calculation methodology. MPM Technologies provided a plant-specific fluence analysis for the RBS capsules. The neutron fluence methodology that was used for benchmarking the RBS unit's capsules complied with the guidelines described in RG 1.190.

(2) Review of Analytic Uncertainty Analysis Identifying Possible Sources of Uncertainty:

The licensee also referenced the EPRI report, "BWR Vessel and Internals Project River Bend 183 Degree Surveillance Capsule Report (BWRVIP-113)" (Reference 6). The report documented an analytical uncertainty analysis performed specifically for the RBS application of the MPM Technologies fluence calculation methodology of Reference 5. Reference 5 also documented the Nine Mile Point licensee's benchmark of the methodology. The licensee determined that the uncertainty for its capsule fluence was 14.6 percent, and the uncertainty for the maximum vessel fluence was 17.4 percent. Regulatory Position 1 of RG 1.190 states that uncertainty in the reactor vessel fluence must be 20 percent or less. Therefore, the NRC staff finds the results of the licensee's analytical uncertainty analysis acceptable.

(3) Benchmark Comparison to an Approved Test Facility:

The NRC staff concluded that the methodology employed at Nine Mile Point was benchmarked acceptably for use in Nine Mile Point-specific applications and is documented in a memorandum from J. Uhle, U.S. Nuclear Regulatory Commission, to R. Laufer, U.S. Nuclear Regulatory Commission, "Nine Mile Point 1, Pressure Vessel Fluence Methodology Review," September 9, 2003 (Reference 7). Part of the basis for this determination was the fact that the methodology was qualified using benchmark data from the Pool Critical Assembly at the Oak Ridge National Laboratory, which meets the intent of RG 1.190. The NRC staff finds that this benchmark applies also to the RBS fluence calculation because the licensee for RBS employed the same methodology.

(4) Plant-Specific Qualification by Comparison to Measured Fluence Values:

In addition to providing the aforementioned analytical uncertainty analysis, the BWRVIP-113 report also provided qualification of the MPM Technologies methodology specifically for RBS. This report documented the comparison of calculated-to-measured (C/M) fluence values. The licensee used two sets of iron and copper wires to collect in-vessel dosimetry capsule fluence measurements. The capsule had been irradiated for a total of 10 EFPY at the time of measurement. The licensee has determined through comparison of its fluence calculation methodology to the measured fluence values that the results had C/M ratios within the required range. Specifically, C/M ratios averaged for the capsule were 0.858. The licensee also referenced a more recent calculation, which included a more geometrically complete model and included additional irradiation history for RBS, and determined that the updated C/M ratio was 0.936 for the Cycle 9 Capsule. The NRC staff finds that these C/M fluence ratios are acceptable. Both measurements are within the uncertainty range of 20 percent, as described by RG 1.190 and are, therefore, acceptable.

Based on the above discussion, the NRC staff finds that the methodology (addressed in Reference 5) that is applied to RBS complies with that described in the RG 1.190. The NRC staff also notes that the licensee's submission has provided information about additional conservatisms that ensure that implementation of the requested relief will continue to ensure adequate protection of public health and safety. These additional conservatisms include the fact that the RPV at RBS contains no circumferential welds in the reactor beltline region, and that the licensee has conservatively applied expected fluence values calculated within the beltline region to the circumferential welds.

The licensee, in its submission, noted that the RBS beltline region contains no circumferential welds. The closest welds are located 7 inches below and 19 inches above the reactor beltline. The licensee conservatively averaged the peak neutron fluence with the fluence calculated for the top of active fuel. Given that, outside the beltline region, the neutron fluence decreases rapidly with increasing distance from the beltline, the NRC staff finds that this conservative application of fluence values is acceptable. The licensee indicated that its conservatively determined fluence estimate for the EOL peak fluence for the circumferential welds is 0.298 x 10^{19} n/cm^2 (E > 1 MeV). The limiting EOL neutron fluence determined by the NRC staff for an RPV fabricated by CB&I is 0.51 x 10^{19} n/cm^2 (E > 1 MeV). The NRC staff therefore concludes, based on the conservative application of fluence estimates to the licensee's circumferential welds, and the acceptability of the fluence estimates for this purpose, that the licensee has met the first criterion of GL 98-05.

3.5.1.2 Evaluation of Circumferential RPV Shell Weld Integrity

The BWRVIP-05 SER evaluated the conditional failure probability of circumferential shell welds for the limiting case of BWR RPVs manufactured by different vendors, including CB&I using the highest mean irradiated RT_{NDT} to determine the limiting case. The RBS beltline does not contain circumferential welds, and the closest two welds are AB, which is approximately 7 inches below the beltline, and AC, which is approximately 19 inches above the beltline region. These two welds, AB and AC, are, therefore, considered the limiting welds for this evaluation. In its submission, the licensee stated that it used a peak fluence value of 0.298 x 10^{19} n/cm^2 (E > 1 MeV) at the inside surface of the limiting welds at EOL (32 EFPY). The corresponding ΔRT_{NDT} value is 109.5 EF for the limiting circumferential shell weld material. Since the RPV at RBS was fabricated by CB&I, the licensee compares the mean irradiated RT_{NDT} for the limiting RBS, circumferential shell weld to that for the limiting CB&I case described in Table 2.6-4 of the BWRVIP-05 SER. As indicated in the licensee's evaluation, the mean RT_{NDT} for the limiting RBS circumferential shell weld is lower than that for the limiting CB&I case. Therefore, the licensee concluded that the conditional failure probability for the RBS circumferential shell welds is bounded by the conditional failure probabilities in the BWRVIP-05 SER through the end of the current license period. Table 3 of this SE provides details of the mean RT_{NDT} for the limiting RBS circumferential shell weld and the staff's bounding criteria for the circumferential shell welds manufactured by CB&I. The NRC staff has verified the licensee's calculated mean RT_{NDT} value for the limiting beltline weld metal.

By comparing the information in the NRC staff's Reactor Vessel Integrity Database with that submitted in the licensee's submission, the NRC staff confirmed that the mean RT_{NDT} of the limiting circumferential shell weld at RBS, is projected to be 30 EF at the end of the current license period of operation (32 EFPY). In this evaluation, the chemistry factor, ΔRT_{NDT}, and mean RT_{NDT} were calculated consistent with the guidelines of RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2. The NRC staff finds the licensee's analysis to be acceptable because it meets the requirements specified in NRC staff's SER for the BWRVIP-05 SER.

The BWRVIP-05 SER provides a limiting conditional failure probability of 2.00 x 10^{-7} per reactor-year for a limiting plant-specific mean RT_{NDT} of 44.5 EF for CB&I fabricated RPVs. The conditional failure probability is the probability of failure if the event were to occur. The LTOP event frequency is the frequency of the event occurring, determined as 10^{-3} per reactor-year in the BWRVIP-05 SER. The vessel failure frequency is the product of conditional failure probability and LTOP frequency. Since the calculated value of mean RT_{NDT} for limiting the circumferential shell weld at RBS is lower than that for the limiting plant-specific case for CB&I fabricated RPV, the RPV failure frequencies of the RBS circumferential shell welds are less than 2.0 X 10^{-10} per reactor-year. Therefore, the conditional failure probability of the RBS RPV circumferential shell welds are bounded by the results obtained in the NRC assessment. A comparison of the data used in the RBS calculation and the NRC staff assessment is provided in Table 3 of this SER.

3.5.2 Criteria 2: Minimizing the Possibility of Low-Temperature Overpressurization (LTOP)

Criterion 2 of GL 98-05 requires that the licensee implement sufficient procedures and/or operator training to ensure that the probability of an LTOP event is minimized. To satisfy the Criterion 2 of GL 98-05, the licensee's submission provided the analysis of the potential high-pressure injection sources, administrative controls, and operator training, all of which are currently in place, that help to minimize the risk of cold overpressurization events.

The licensee in its request, as supplemented, provided its basis for complying with the requirements of Criterion 2 of GL 98-05. It is summarized below:

1. Most high-pressure injection sources that could cause LTOP events are prevented by interlocks, plant conditions, and/or administrative controls.

2. The plant procedures ensure sufficient temperature margin during operation of the condensate pumps to prevent exceeding the reactor's P/T technical specification limits.

3. Numerous indications exist that would inform plant personnel that condensate pump injection was occurring in such a manner that an LTOP event could occur.

4. With regard to CRD pumps, operators closely monitor reactor water level, pressure, and temperature during cold shutdown conditions to ensure sufficient time to regain control in the event that abnormalities occur in the CRD flow.

5. The safety from LTOP events is assured by closely monitoring reactor parameters that would indicate an impending cold overpressurization.

6. The plant procedures require that reactor operators monitor and maintain reactor temperature, pressure, and level within bands of prescribed setpoints during all modes of operation. When any of these parameters fall outside of prescribed setpoints, the reactor operator is required to inform the SRO, who is then responsible for ensuring that mitigative actions are taken to return plant parameters to acceptable values. The required procedures, are reinforced in both classroom and simulator training.

Based on its review of the licensee's evaluation of potential injection sources that would cause an LTOP event, the NRC staff finds that it is consistent with the industry response to the NRC staff's evaluation of the BWRVIP-05 report. In addition, it also addresses the NRC staff's concern, noted in the BWRVIP-05 SER, that CRD and condensate pumps could cause conditions that could lead to LTOP events (Reference 2). Further, the NRC's inspections and audits encompass licensee's procedures and processes discussed above, on periodic basis, for proper implementation. The licensee's performance in this regard, is monitored and corrective actions are mandated as necessary.

Based on the above discussion, the NRC staff finds that the licensee has adequately demonstrated that it has implemented operator training and established procedures that limit the frequency of LTOP events to the amount specified in the BWRVIP-05 SER at RBS. Therefore, the NRC staff finds that licensee has complied with Criterion 2 of GL 98-05 requirements.

TABLE 3

RBS - RPV Shell Weld Information: Bounding Circumferential Shell Weld

Parameter Description	RBS - Shell Bounding Beltline 32 EFPY Bounding Comparative Parameters (Bounding Circ. Welds)	U.S. NRC Limiting 32 EFPY Bounding CB&I Vessel Parameters SER Table 2.6-4
Neutron fluence at the end of the requested relief period (Peak Surface Fluence- E > 1 MeV)	0.298×10^{19} n/cm^2	0.51×10^{19} n/cm^2
Initial (unirradiated) reference temperature (RT_{NDT})	-60 EF	-65 EF
Weld Chemistry Factor (CF)	134.5	134.9
Weld Copper content %	0.10	0.10
Weld Nickel content %	0.95	0.99
Increase in reference temperature (ΔRT_{NDT}),	90 EF	109.5 EF
Mean RT_{NDT}	30 EF	44.5 EF

4.0 CONCLUSIONS

The NRC staff has reviewed the licensee's submittal and finds that the licensee has demonstrated that the appropriate criteria in GL 98-05 and the BWRVIP-05 SER have been satisfied regarding permanent relief (i.e., for the remaining portion of the current 40 year license period) from ISI requirements of the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, for the volumetric examination of the RPV circumferential shell welds.

The NRC staff finds that the licensee's alternative to the examination requirements for circumferential RPV shell welds for the remaining portion of the current 40 year license period, is consistent with the information contained in NRC GL 98-05 and provides an acceptable level of quality and safety. Accordingly, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative examination for RBS is authorized.

Additional requirements of the ASME Code, Section XI, for which relief has not been specifically requested remain applicable, including third party reviews by the Authorized Nuclear Inservice Inspector.

5.0 REFERENCES

1. Electric Power Research Institute, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," TR-105697, Palo Alto, California, September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, May 16, June 4, June 13, and December 18, 1997, and January 13, 1998.

2. U.S. Nuclear Regulatory Commission, "Final Safety Evaluation of the BWR Vessels and Internals Project BWRVIP-05," July 28, 1998.

3. U.S. Nuclear Regulatory Commission, Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examinations Requirements on Reactor Pressure Vessel Circumferential Shell Welds," November 10, 1998

4. U.S. Nuclear Regulatory Commission, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Regulatory Guide (RG) 1.190, March 2001.

5. Manahan, M. P., "Benchmarking of Nine Mile Point Unit 1 and Unit 2 Neutron Transport Calculations (MPM-402781)," Revision 1, State College, Pennsylvania, September 2003.

6. Electric Power Research Institute, "BWR Vessel and Internals Project River Bend 183 Degree Surveillance Capsule Report (BWRVIP-113)," TR-1003345, Palo Alto, California, June 2003.

7. U.S. Nuclear Regulatory Commission, memorandum from J. Uhle to R. Laufer, "Nine Mile Point 1, Pressure Vessel Fluence Methodology Review," September 9, 2003.

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