Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
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J.E. Pollock
Site Vice President
Administration

July 8, 2008

Re: Indian Point Unit 2 and 3
Docket Nos. 50-247 and 50-286

NL-08-096

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

SUBJECT: Request For Relief To Extend The Unit 2 and 3 Inservice Inspection Interval For The Reactor Vessel Weld Examination And Request For License Amendment For Submittal of ISI Information and Analyses

Reference: Ho K. Nieh, NRC, NRR Letter to Gordon Bischoff, WOG regarding Final Safety Evaluation For PWROG Topical Report WCAP-16168-NP, Revision 2, (TAC NO. MC9768), dated May 8, 2008.

Dear Sir or Madam:

Entergy Nuclear Operations, Inc. (Entergy) is submitting Relief Request No. 76 (RR-76) (Enclosure 1) for Indian Point Unit No. 2 (IP2) and Relief Request No. 3-43(I) (RR-3-43(I)) (Enclosure 2) for Indian Point Unit No. 3 (IP3). These relief requests are for the Third 10-year Inservice Inspection (ISI) Interval and will also apply to the fourth 10-year ISI Interval for the Reactor Vessel (RV) Weld examinations.

The NRC approved WCAP-16168-NP-A, Revision 2, "Risk- Informed Extension of The Reactor Vessel In-Service Inspection Interval," in the above referenced letter. This WCAP provides for extension of the inservice inspection interval for certain pressure retaining welds in the reactor vessel from 10 to 20 years. Entergy proposes to implement this extended inservice inspection interval for IP2 and IP3. The plant specific information identified by the above letter as needed to support this request is in enclosures 1 and 2. Entergy has concluded that the proposed alternative provides an acceptable level of quality and safety. The relief is requested under the provisions of 10CFR 50.55a(a)(3)(i).

As required by the referenced letter, Entergy is requesting an amendment to the IP2 and IP3 licenses that will require that the information and analyses requested in the final rule for 10 CFR 50.61a, Section (e) or, prior to issuance, the proposed rule (72 FR 56275) for 10 CFR 50.61a, Section (e) be submitted within one year of completing each of the ASME Code,

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NRR

Section XI, Category B-A and B-D RV weld inspections. Entergy has evaluated the proposed change in accordance with 10 CFR 50.91 (a)(1) using the criteria of 10 CFR 50.92 (c) and determined that this proposed change involves no significant hazards considerations. The proposed change and evaluation are contained in Attachment 1. A copy of this application, the attachment and the enclosures to this letter are being submitted to the designated New York State official.

Entergy requests approval of the relief requests and license amendment by February 2009, to support IP3 Refueling Outage (RFO) – 3R15.

There are no new commitments identified in this submittal. If you have any questions or require additional information, please contact Mr. Robert Walpole, Licensing Manager at 914-734-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 7-8-08.

Very truly yours,

Patrick W. Conroy for
J. E. Pollock
Site Vice President
Indian Point Energy Center

Attachment 1. Proposed License Amendment Regarding ASME Relief Request Information and Analysis Per 10 CFR 50.91 and 10 CFR 50.92

Enclosure 1. IP2 Request For Relief to Extend the Third 10-year Reactor Vessel Inservice Inspection Interval Relief Request No: RR-76

2. IP3 Request For Relief to Extend the Third 10-year Reactor Vessel Inservice Inspection Interval Relief Request No: RR-3-43(I)

cc:
Mr. John P. Boska, Senior Project Manager, NRC NRR DORL
Mr. Samuel J. Collins, Regional Administrator, NRC Region I
NRC Resident Inspector's Office Indian Point
Mr. Paul Eddy, New York State Department of Public Service
Mr. Paul D. Tonko, President NYSERDA

ATTACHMENT 1 TO NL-08-096

PROPOSED LICENSE AMENDMENT REGARDING ASME RELIEF REQUEST

INFORMATION AND ANALYSIS PER 10 CFR 50.91 AND 10 CFR 50.92

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3
DOCKET NOS. 50-247 AND 50-286

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90 and 10 CFR 50.91(a)(5), Entergy Nuclear Operations, Inc (Entergy) hereby requests an amendment to the Indian Point 2 (IP2) and Indian Point 3 (IP3) Licenses. Entergy has requested a Reactor Vessel Inservice Inspection Relief Request for each unit based on the NRC approved Topical Report WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval." The NRC safety evaluation report approving the WCAP required Licensees requesting the relief to submit a request to amend the license. The purpose of this request is to comply with that requirement.

2.0 PROPOSED CHANGE

The proposed change to IP2 License will add item (4) to Section 2.C that will read as follows:

> "(4) The following condition relates to the Relief Request to extend the Reactor Vessel inservice inspection interval: Provide the NRC with the information and analysis requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275, prior to issuance of the final 10 CFR 50.61a) following completion of each ASME Code, Section XI, Category B-A and B-D Reactor Vessel weld inspection. The information must be submitted within one year of the inspection."

The proposed change to the IP3 License will revise Section 2.C from:

"(3) (DELETED)	Amdt. 205 2-27-01"

To

> "(3) The following condition relates to the Relief Request to extend the Reactor Vessel inservice inspection interval: Provide the NRC with the information and analysis requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275, prior to issuance of the final 10 CFR 50.61a) following completion of each ASME Code, Section XI, Category B-A and B-D Reactor Vessel weld inspection. The information must be submitted within one year of the inspection."

3.0 BACKGROUND

The Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) WCAP-16168-NP, Revision 1, "RISK- INFORMED EXTENSION OF THE REACTOR VESSEL IN-SERVICE INSPECTION INTERVAL," to the U.S. Nuclear Regulatory Commission (NRC) staff by letter dated January 26, 2006 and supplemented by letter dated June 8, 2006. PWROG letter dated October 16, 2007 submitted TR WCAP-16168-NP , Revision 2, and responses to the NRC staff's request for additional information (RAI) for NRC staff review by. An NRC draft safety evaluation (SE) regarding approval of TR WCAP-16168-NP, Revision 2, was provided to the PWROG for review and comments by letter dated March 6, 2008. Comments were provided by the PWROG by letter dated March 31, 2008.

The NRC issued a final safety evaluation (SE) and approval of TR WCAP-16168-NP, Revision 2 by letter dated May 8, 2008. The NRC staff's disposition of PWROG comments on the draft SE are discussed in an attachment to the May 8, 2008 letter. The SE attached to the May 8, 2008 letter identifies the information requirements to be included in the relief request and requires that Licensees submit the information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275, prior to issuance of the final 10 CFR 50.61a) within one year of completing each ASME Code, Section XI, Category B-A and B-D Reactor Vessel weld inspection. To administratively control the submission of this information the SE also requires that "Licensees that do not implement 10 CFR 50.61a must amend their licenses to require that the information and analyses requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275, prior to issuance of the final 10 CFR 50.61a) will be submitted for NRC staff review and approval. The amendment to the license shall be submitted at the same time as the request for alternative." Entergy is not implementing 10 CFR 50.61a since the rule is not final. This amendment request implements the requirement to submit a license amendment request at the time of submitting a request for the alternative.

4.0 TECHNICAL ANALYSIS

The addition of a License condition to require the submission of the information and analysis requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275, prior to issuance of the final 10 CFR 50.61a) following completion of each ASME Code, Section XI, Category B-A and B-D Reactor Vessel weld inspection according to the criteria of 10 CFR 50.92, is an administrative change with no effect on the public safety. The change provides the NRC assurance that Entergy will submit defined information and analyses to the NRC every time that a specific inservice inspection is done. Submission of the information and analyses can have no effect on the consequences of an accident or the probability of an accident. The submission has no effect on the manner in which the plant or its equipment is operated, it has no effect on the programs and processes for training personnel and for personnel to operate equipment, and it has no effect on the manner in which accident analyses are performed. The submission of information cannot create the possibility of a new or different kind of accident from any accident previously evaluated for the same reasons. The proposed change cannot have a significant effect on the margin of safety because it is not related to any margin of safety. The relief requests to extend the ISI from 10 to 20 years is separate from this License change and are approved independently.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (Entergy) has evaluated the safety significance of the proposed change regarding the addition of a License condition to submit the information and analysis requested in Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275, prior to issuance of the final 10 CFR 50.61a) following completion of each ASME Code, Section XI, Category B-A and B-D Reactor Vessel weld inspection according to the criteria of 10 CFR 50.92, "Issuance of Amendment". Entergy has determined that the subject change does not involve a Significant Hazards Consideration as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No. The proposed change will revise the license to require the submission of information and analyses to the NRC following completion of each ASME Code, Section XI, Category B-A and B-D Reactor Vessel weld inspection. The extension of the ISI from 10 to 20 years is being evaluated as part of the relief request independent from the license change. Submission of the information and analyses can have no effect on the consequences of an accident or the probability of an accident because the submission of information is not related to the operation of the plant or any equipment, the programs and procedures used to operate the plant, or the evaluation of accidents. The submittal of information and analyses provides the opportunity for the NRC to independently assess the information and analyses.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No. The proposed change will only affect the requirement to submit information and analyses when specified inspections are performed. There are no changes to plant equipment, operating characteristics or conditions, programs and procedures or training. Therefore, there are no potential new system interactions or failures that could create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No. The proposed change will revise the license to require the submission of information and analyses to the NRC following completion of each ASME Code, Section XI, Category B-A and B-D Reactor Vessel weld inspection which does not affect any Limiting Conditions for Operation used to establish the margin of safety. The requirement to submit information and analyses is an administrative tool to assure the NRC has the ability to independently review information developed by the Licensee. The proposed change does not involve a significant reduction in the margin of safety.

Based on the above, Entergy Nuclear Operations, Inc. concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 <u>Applicable Regulatory Requirements / Criteria</u>

The proposed change has been reviewed to evaluate the potential effect on regulatory requirements and criteria. There are no rules and regulations requiring the submittal of information and analyses to NRC regarding NRC ASME Code, Section XI, Category B-A and B-D Reactor Vessel weld inspection. The information and analyses of Section (e) of the proposed 10 CFR 50.61a defines the requirements for verifying that the pressurized thermal shock screening criteria of the proposed rule are applicable to the reactor vessel. The final rule will be the same or modified as a result of comments. The amendment is the administrative means chosen by the NRC staff to obtain this information.

5.3 Environmental Considerations

The proposed change to the IP2 and IP3 Licenses regarding submittal of the information required by Section (e) of the final 10 CFR 50.61a (or the proposed 10 CFR 50.61a, given in 72 FR 56275, prior to issuance of the final 10 CFR 50.61a) does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

The proposed change is consistent with the requirements of the NRC safety evaluation in Reference 1. There is no existing precedent due to the short time frame that has passed since the NRC approval.

7.0 REFERENCES

1. Ho K. Nieh, NRC, NRR Letter to Gordon Bischoff, WOG regarding Final Safety Evaluation For PWROG Topical Report WCAP-16168-NP, Revision 2, (TAC NO. MC9768), dated May 8, 2008.

ENCLOSURE 1 TO NL-08-096

IP2 REQUEST FOR RELIEF TO EXTEND THE THIRD 10-YEAR REACTOR VESSEL INSERVICE INSPECTION INTERVAL RELIEF REQUEST NO: RR-76

(9 Pages)

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

Indian Point Unit 2

Third 10-year ISI Interval

Relief Request No: RR-76

Reactor Vessel Inservice Inspection Interval Extension

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)
-Alternative Provides Acceptable Level of Quality and Safety-

1. ASME Code Component(s) Affected

The affected component is the Indian Point Unit 2 (IP2) reactor vessel (21RV), specifically the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code Section XI (Reference 1) examination categories and item numbers covering examinations of the reactor vessel (RV). These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME BPV, Code Section XI.

Examination Category	Item No.	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional Shell Welds
B-A	B1.30	Shell-to-Flange Weld
B-A	B1.40	Head-to-Flange Weld
B-A	B1.50	Repair Welds
B-A	B1.51	Beltline Region
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inner Radius Areas

(Throughout this request the above examination categories are referred to as "the subject examinations" and the ASME BPV Code, Section XI, is referred to as "the Code.")

2. Applicable Code Edition and Addenda

ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Code 1989 Edition, No Addenda.

3. Applicable Code Requirement

IWB-2412, Inspection Program B, requires volumetric examination of essentially 100% of reactor pressure vessel pressure retaining welds identified in Table IWB-2500-1 once each ten year interval. The IP2 third inspection interval was scheduled to end on or before December 31, 2006. This interval was first extended by one refueling outage via relief request number 73 (Reference 2). The third interval was then extended by a second one refueling outage to 2010 via relief request number RR-06 (Reference 3).

4. Reason for Request

An alternative is requested from the requirement of IWA-2412, Inspection Program B, that volumetric examination of essentially 100% of reactor pressure vessel retaining welds, Examination Categories B-A and B-D welds, be performed once each ten-year interval. Extension of the inspection interval for Examination Category B-A and B-D welds from 10 years to up to 20 years will result in a reduction in man-rem exposure and examination costs.

5. Proposed Alternative and Basis for Use

Indian Point Unit 2 proposes to defer completion of the ASME Code required volumetric examination of the Reactor Pressure Vessel full penetration pressure retaining Category B-A and B-D welds for the third inservice inspection interval until 2012 and to perform the fourth interval inservice inspection on a twenty-year inspection interval, instead of the currently required ten-year inspection interval. Therefore, the fourth interval inservice inspection is proposed to be completed by 2032. These dates are consistent with the information provided to the Staff in PWR Owners Group letter OG-06-356 (Reference 4).

In accordance with 10 CFR 50.55a(a)(3)(i), an alternate inspection interval is requested on the basis that the current inspection interval can be extended based on a negligible change in risk by satisfying the risk criteria specified in Regulatory Guide 1.174 (Reference 5).

The methodology used to demonstrate the acceptability of extending the third and fourth inspection intervals for Category B-A and B-D welds based on a negligible change in risk is contained in WCAP-16168-NP-A, Revision 2 (Reference 6). This methodology was used to develop a pilot plant analysis for Westinghouse, Combustion Engineering, and Babcock and Wilcox reactor vessel designs and is an extension of the work that was performed as part of the NRC PTS Risk Re-Evaluation (Reference 7). The critical parameters for demonstrating that this pilot plant analysis is applicable on a plant specific basis, as identified in WCAP-16168-NP-A, Revision 2, are identified in Table 1. By demonstrating that each plant specific parameter is bounded by the corresponding pilot plant parameter, the application of the methodology to the Indian Point Unit 2 reactor vessel is acceptable as shown in Table 1 below.

Table 1	Critical Parameters for Application of Bounding Analysis		
Parameter	**Pilot Plant Basis**	**Plant Specific Basis**	**Additional Evaluation Required?**
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are applicable	NRC PTS Risk Study (Reference 7)	PTS Generalization Study (Reference 8)	No
Through Wall Cracking Frequency	WCAP-16168-NP-A, Revision 2, 1.76E-08 Events per year (Reference 6)	3.08E-07 Events per year (Calculated per Reference 7)	Yes (See discussion below)
Frequency and Severity of Design Basis Transients	WCAP-16168-NP-A, Revision 2, Bounded by 7 cooldowns per year (Reference 6)	Bounded by 7 cooldowns per year	No
Cladding Layers (Single/Multiple)	WCAP-16168-NP-A, Revision 2, Single Layer (Reference 6)	Single Layer	No

As shown in Table 1, the pilot plant through wall cracking frequency (TWCF) value is not bounding of the IP2 value. Therefore, an additional evaluation was performed as documented in RR–76 Attachment 1 "Additional Anayses for the TWCF Parameter" to this enclosure. The Westinghouse pilot plant change in risk analysis in WCAP-16168-NP-A, Revision 2 was performed at 60 EFPY. To bound IP2 the Westinghouse pilot plant was reevaluated at a value of EFPY (Condition B) that is well beyond 60 EFPY. The change in risk associated with extending the ISI interval at Condition B is 3.51E-08 events per year. This change in risk is acceptably small per Regulatory Guide 1.174 (Reference 5). The TWCF value for the Westinghouse pilot plant at Condition B is 9.33E-07 events per year. This TWCF value is greater than the IP2 TWCF value and therefore the pilot plant analysis is bounding of IP2 for this parameter.

Additional information relative to the Indian Point Unit 2 reactor vessel inspections is provided in Table 2. This information confirms that satisfactory examinations have been performed on the Indian Point Unit 2 reactor vessel.

Table 2	**Additional Information Pertaining to Reactor Vessel Inspection**
Inspection methodology:	Regulatory Guide 1.150 (Reference 9)
Number of past inspections:	A minimum of 2 inspections have been performed on each weld.
Number of indications found:	One reportable indication was found during the first ten-year reactor vessel examination in 1984. This indication was re-examined in 1987 and 1995 using more advanced techniques and found to be acceptable per ASME Section XI IWB-3500. Any recordable indications have been acceptable per ASME Section XI IWB-3500.
Proposed inspection schedule for balance of plant life:	The third interval inservice inspection was scheduled to be performed in 2006. This inspection was deferred and is currently scheduled to be completed by 2010. The third interval inservice inspection is proposed to be completed by 2012. If applicable, the fourth interval inservice inspection will be completed by 2032.

The information in Table 3 is identified in WCAP-16168-NP-A, Revision 2, as additional information to be provided relative to the TWCF calculation.

Table 3 Details of TWCF Calculation

Inputs								
Reactor Coolant System Temperature, T_{RCS}[°F]:			523		T_{wall} [inches]:			8.84
#	Region/Component Description	Material	Cu [wt%]	Ni [wt%]	P [wt%]	Mn [wt%]	Un-Irradiated $RT_{NDT(u)}$ [°F]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]
1	Intermediate Plate	A 302BM	0.19	0.65	0.010	1.45	34	2.38
2	Intermediate Plate	A 302BM	0.17	0.46	0.014	1.45	21	2.38
3	Intermediate Plate	A 302BM	0.25	0.60	0.011	1.45	21	2.38
4	Lower Plate	A 302BM	0.20	0.66	0.010	1.45	20	2.38
5	Lower Plate	A 302BM	0.19	0.60	0.011	1.45	-20	2.38
6	Int./Lower Circ Weld	Linde 1092	0.19	1.01	0.010	1.63	-56	2.38
7	Inter. Axial Weld	Linde 1092	0.21	1.01	0.021	1.63	-56	1.62
8	Inter. Axial Weld	Linde 1092	0.21	1.01	0.021	1.63	-56	1.62
9	Inter. Axial Weld	Linde 1092	0.21	1.01	0.021	1.63	-56	0.83
10	Lower Axial Weld	Linde 1092	0.21	1.01	0.021	1.63	-56	1.32
11	Lower Axial Weld	Linde 1092	0.21	1.01	0.021	1.63	-56	1.32

Outputs						
Methodology Used to Calculate ΔT_{30}:		NUREG-1874				
	Controlling Material Region # (From Above)	RT_{MAX-XX} [R]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]	ϕ (flux)	ΔT_{30} [°F]	$TWCF_{95-XX}$
Axial Weld – AW	7, 8	705.85	1.62	8.55E+09	302.16	1.50E-07
Circumferential Weld - CW	3	709.42	2.38	1.26E+10	228.73	5.56E-11
Plate – PL	3	709.42	2.38	1.26E+10	228.73	3.05E-09
$TWCF_{95-TOTAL}$ ($\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW} + \alpha_{FO}TWCF_{95-FO}$):						3.08E-07

6. Duration of Proposed Alternative

This request is applicable to Entergy's inservice inspection program for the third and fourth inspection intervals for Indian Point Unit 2.

7. References

1. *ASME Boiler and Pressure Vessel Code,* Section XI, 1989 Edition no Addenda, American Society of Mechanical Engineers, New York.
2. USNRC to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Relief Request (RR) No. 73 (TAC No. MC7306)," February 22, 2006.
3. USNRC to Entergy, "Indian Point Nuclear Generating Unit No. 2 – Relief Request (RR) No. RR-06 (TAC No. MC4701)," October 29, 2007
4. OG-06-356, "Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval." MUHP 5097-99, Task 2059," October 31, 2006.
5. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.
6. WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval," June 2008.
7. NUREG-1874, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)," 3/1/07.
8. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004.
9. NRC Regulatory Guide 1.150, Revision 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," February 1983.

RR–76 Attachment 1: Additional Analysis for the TWCF Parameter

Purpose

The change in risk analysis for the Westinghouse pilot plant in WCAP-16168-NP-A, Revision 2, Beaver Valley Unit 1, was performed at 60 EFPY with the intention of bounding the embrittlement of all Westinghouse plants as determined using the through wall cracking frequency (TWCF) correlation in Reference 1. As shown in Table 1 of RR-76, the TWCF value calculated for Indian Point Unit 2 is not bounded by the Westinghouse pilot plant value for 60 EFPY. However, the TWCF calculated for the Westinghouse pilot plant at the EFPY for Condition B ("Ext-Bb") in Reference 2 is 9.33E-07 events per year. This value is bounding of the Indian Point Unit 2 TWCF value. Condition B is well beyond 60 EFPY so that the embrittlement levels will provide a TWCF that exceeds the Regulatory Guide 1.174 (Reference 3) risk limit of 1.0E-07 events per year. The purpose of the analysis documented in this attachment is to perform the change in risk calculation for the Westinghouse pilot plant at the EFPY for condition B. If the change in risk at this EFPY is acceptable, then the Westinghouse pilot plant bounds Indian Point Unit 2 for the TWCF parameter.

Method Discussion

The analysis performed in this attachment is consistent with that performed for the Westinghouse pilot plant in Reference 2. These calculations are performed at the EFPY for Condition B. All inputs are consistent with those used in Reference 2 with the exception that the fluence values in the FAVPFM input file correspond to the EFPY for Condition B rather than 60 EFPY. This input file is consistent with that used for Condition B in the NRC PTS Risk Re-Evaluation (Reference 1).

Results

Consistent with Reference 2, the FAVOR Code evaluation was run for each analysis for a number of vessel simulations required to obtain a stable solution. A summary of failure frequency results is included in Table 1 for the FAVPOST output for the 10 Year ISI only and 10 Year ISI Interval runs for Beaver Valley Unit 1.

Table 1: Failure Frequency Results (Events per Year)	
Case	Beaver Valley Unit 1 – Condition B
10 Year ISI Only (Mean Value)	5.08E-07
10 Year ISI Only (Standard Error)	8.23E-09
10 Year Interval (Mean Value)	5.09E-07
10 Year Interval (Standard Error)	9.69E-09

For delta risk calculations, the difference is typically taken between mean values. Consistent with Reference 2, an upper and lower bound were determined for the "10 Year ISI Interval" and "10 Year ISI Only" cases. The Upper Bound is calculated by adding two times the Standard Error to the "10 Year ISI Only" mean failure frequencies. The Lower Bound is calculated by subtracting two times the Standard Error from the "10 Year ISI Interval" mean failure frequencies. The change in failure frequency is conservatively calculated by taking the difference between these upper and lower bounds at the last vessel simulation. Table 2 displays the change in failure frequency results at the last vessel simulation. Figure 1 shows the change in failure frequencies calculated for different numbers of iterations or vessel simulations.

Table 2: Change in Failure Frequency Results – 10 Year ISI Interval and 10 Year ISI Only (Events per Year)	
Case	Beaver Valley Unit 1 – Condition B
Upper Bound	5.25E-07
Lower Bound	4.90E-07
Delta Risk (Mean Values)	-7.80E-10
Bounding Delta Risk	3.51E-08

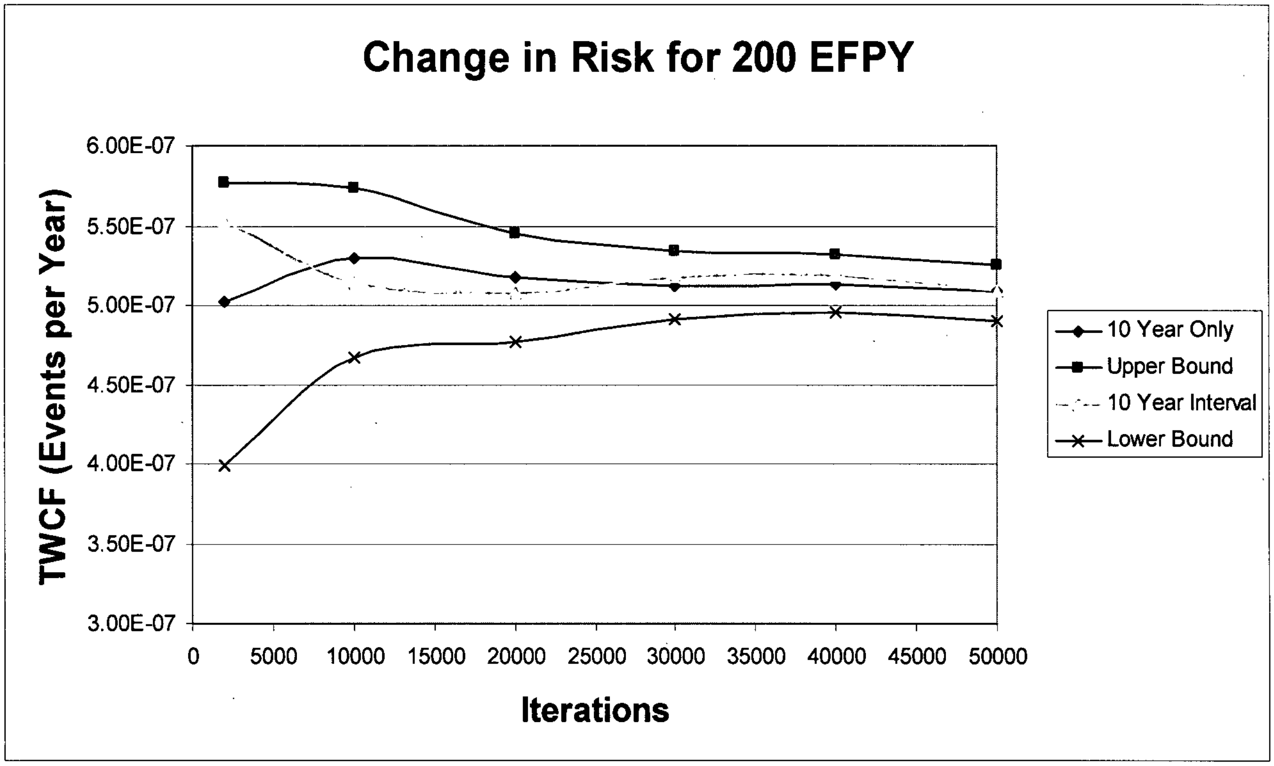

Figure 1: Comparison of Results for Beaver Valley Unit 1 10 Year ISI Interval and 10 Year ISI Only

Conclusions

In Regulatory Guide 1.174 (Reference 3), the acceptable change in risk was specified as 1.0E-7 events per year. As seen in Table 2, the bounding change in risk (delta risk) is below the acceptable change. Therefore, the conclusions of Reference 2 remain valid for the Westinghouse pilot plant (Beaver Valley Unit 1) at Condition B and for Indian Point Unit 2 at 60 EFPY.

References

1. NUREG-1874, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)," 3/1/07.
2. WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," June 2008.
3. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.

ENCLOSURE 2 TO NL-08-096

IP3 REQUEST FOR RELIEF TO EXTEND THE THIRD 10-YEAR REACTOR VESSEL INSERVICE INSPECTION INTERVAL RELIEF REQUEST NO: RR-3-43(I)

(9 Pages)

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

Indian Point Unit 3
Third 10-year ISI Interval
Relief Request No: 3-43(I)
Reactor Vessel Inservice Inspection Interval Extension
Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)
-Alternative Provides Acceptable Level of Quality and Safety-

1. ASME Code Component(s) Affected

The affected component is the Indian Point Unit 3 (IP3) reactor vessel, specifically the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code Section XI (Reference 1) examination categories and item numbers covering examinations of the reactor vessel (RV). These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME BPV Code, Section XI.

Examination Category	Item No.	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional Shell Welds
B-A	B1.30	Shell-to-Flange Weld
B-A	B1.40	Head-to-Flange Weld
B-A	B1.50	Repair Welds
B-A	B1.51	Beltline Region
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inner Radius Areas

(Throughout this request the above examination categories are referred to as "the subject examinations" and the ASME BPV Code, Section XI, is referred to as "the Code.")

2. Applicable Code Edition and Addenda

ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Code 1989 Edition, No Addenda.

3. Applicable Code Requirement

IWB-2412, Inspection Program B, requires volumetric examination of essentially 100% of reactor pressure vessel pressure retaining welds identified in Table IWB-2500-1 once each ten year interval. The IP3 third inspection interval is scheduled to end on or before July 21, 2009.

4. Reason for Request

An alternative is requested from the requirement of IWA-2412, Inspection Program B, that volumetric examination of essentially 100% of reactor pressure vessel retaining welds, Examination Categories B-A and B-D welds, be performed once each ten-year interval. Extension of the inspection interval for Examination Category B-A and B-D welds from 10 years to up to 20 years will result in a reduction in man-rem exposure and examination costs.

5. Proposed Alternative and Basis for Use

Indian Point Unit 3 proposes to defer the completion of ASME Code required volumetric examination of the Reactor Pressure Vessel full penetration pressure retaining Category B-A and B-D welds for the third interval inservice inspection until 2015 and to perform the fourth interval inservice inspection on a twenty-year inspection interval, instead of the currently required ten-year inspection interval. Therefore, the fourth interval inservice inspection is proposed to be completed by 2035. These dates are consistent with the information provided to the Staff in PWR Owners Group letter OG-06-356 (Reference 2).

In accordance with 10 CFR 50.55a(a)(3)(i), an alternate inspection interval is requested on the basis that the current inspection interval can be extended based on a negligible change in risk by satisfying the risk criteria specified in Regulatory Guide 1.174 (Reference 3).

The methodology used to demonstrate the acceptability of extending the third and forth inspection intervals for Category B-A and B-D welds based on a negligible change in risk is contained in WCAP-16168-NP-A, Revision 2 (Reference 4). This methodology was used to develop a pilot plant analysis for Westinghouse, Combustion Engineering, and Babcock and Wilcox reactor vessel designs and is an extension of the work that was performed as part of the NRC PTS Risk Re-Evaluation (Reference 5). The critical parameters for demonstrating that this pilot plant analysis is applicable on a plant specific basis, as identified in WCAP-16168-NP-A, Revision 2, are identified in Table 1. By demonstrating that each plant specific parameter is bounded by the corresponding pilot plant parameter, the application of the methodology to the Indian Point Unit 3 reactor vessel is acceptable as shown in Table 1 below.

Table 1	Critical Parameters for Application of Bounding Analysis		
Parameter	**Pilot Plant Basis**	**Plant Specific Basis**	**Additional Evaluation Required?**
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are applicable	NRC PTS Risk Study (Reference 5)	PTS Generalization Study (Reference 6)	No
Through Wall Cracking Frequency	WCAP-16168-NP-A, Revision 2, 1.76E-08 Events per year (Reference 4)	4.12E-07 Events per year (Calculated per Reference 5)	Yes (See discussion below)
Frequency and Severity of Design Basis Transients	WCAP-16168-NP-A, Revision 2, Bounded by 7 cooldowns per year (Reference 4)	Bounded by 7 cooldowns per year	No
Cladding Layers (Single/Multiple)	WCAP-16168-NP-A, Revision 2, Single Layer (Reference 4)	Single Layer	No

As shown in Table 1, the pilot plant through wall cracking frequency (TWCF) value is not bounding of the IP3 value. Therefore, an additional evaluation was performed as documented in RR–3–43(I) Attachment 1 "Additional Analysis for the TWCF Parameter" to this enclosure. The Westinghouse pilot plant change in risk analysis in WCAP-16168-NP-A, Revision 2 was performed at 60 EFPY. To bound IP3 the Westinghouse pilot plant was reevaluated at a value of EFPY (Condition B) that is well beyond 60 EFPY. The change in risk associated with extending the ISI interval at Condition B is 3.51E-08 events per year. This change in risk is acceptably small per Regulatory Guide 1.174 (Reference 3). The TWCF value for the Westinghouse pilot plant at Condition B is 9.33E-07 events per year. This TWCF value is greater than the IP3 TWCF value and therefore the pilot plant analysis is bounding of IP3 for this parameter.

Additional information relative to the Indian Point Unit 3 reactor vessel inspections is provided in Table 2. This information confirms that satisfactory examinations have been performed on the Indian Point Unit 3 reactor vessel.

Table 2 Additional Information Pertaining to Reactor Vessel Inspection

Inspection methodology:	Regulatory Guide 1.150 (Reference 7)
Number of past inspections:	A minimum of 2 inspections have been performed on each weld.
Number of indications found:	No reportable indications have been found in the reactor vessel. Any recordable indications have been acceptable per ASME Section XI IWB-3500.
Proposed inspection schedule for balance of plant life:	The third interval inservice inspection is currently scheduled for Spring of 2009. The third interval inservice inspection is proposed to be completed by 2015. If applicable, the fourth interval inservice inspection will be completed by 2035.

The information in Table 3 is identified in WCAP-16168-NP-A, Revision 2, as additional information to be provided relative to the TWCF calculation.

Table 3	**Details of TWCF Calculation- Indian Point Unit 3**							
Inputs								
	Reactor Coolant System Temperature, T_{RCS}[°F]:			538		T_{wall} [inches]:		8.92
#	Region/Component Description	Material	Cu [wt%]	Ni [wt%]	P [wt%]	Mn [wt%]	Un-Irradiated $RT_{NDT(u)}$ [°F]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]
1	Intermediate Plate	A 302BM	0.200	0.500	0.010	1.45	5	1.95
2	Intermediate Plate	A 302BM	0.220	0.530	0.015	1.45	-4	1.95
3	Intermediate Plate	A 302BM	0.200	0.490	0.011	1.45	17	1.95
4	Lower Plate	A 302BM	0.190	0.470	0.012	1.45	49	1.95
5	Lower Plate	A 302BM	0.220	0.520	0.011	1.45	-5	1.95
6	Lower Plate	A 302BM	0.240	0.520	0.012	1.45	74	1.95
7	Int./Lower Circ Weld	Linde 1092	0.221	0.732	0.023	1.63	-54	1.95
8	Inter. Axial Weld	Linde 1092	0.192	1.007	0.012	1.63	-56	1.55
9	Inter. Axial Weld	Linde 1092	0.192	1.007	0.012	1.63	-56	1.55
10	Inter. Axial Weld	Linde 1092	0.192	1.007	0.012	1.63	-56	0.909
11	Lower Axial Weld	Linde 1092	0.192	1.007	0.012	1.63	-56	1.39
12	Lower Axial Weld	Linde 1092	0.192	1.007	0.012	1.63	-56	1.39
13	Lower Axial Weld	Linde 1092	0.192	1.007	0.012	1.63	-56	1.95
Outputs								
	Methodology Used to Calculate ΔT_{30}:			NUREG-1874				
		Controlling Material Region # (From Above)	RT_{MAX-XX} [R]	Fluence [10^{19} Neutron/cm^2, E>1 MeV]	ϕ (flux)	ΔT_{30} [°F]	$TWCF_{95-XX}$	
Axial Weld – AW		6	710.82	1.39	7.33E+09	177.13	2.02E-07	
Circumferential Weld - CW		6	720.23	1.95	1.03E+10	186.54	1.51E-10	
Plate – PL		6	720.23	1.95	1.03E+10	186.54	5.66E-09	
$TWCF_{95-TOTAL}$ ($\alpha_{AW}TWCF_{95-AW} + \alpha_{PL}TWCF_{95-PL} + \alpha_{CW}TWCF_{95-CW} + \alpha_{FO}TWCF_{95-FO}$):							4.12E-07	

6. Duration of Proposed Alternative

This request is applicable to Entergy's inservice inspection program for the third and fourth inspection intervals for Indian Point Unit 3.

7. References

1. ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition with the 1989 Addenda up to and including the 2004 Edition with the 2005 Addenda, American Society of Mechanical Engineers, New York.
2. OG-06-356, "Plan for Plant Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval." MUHP 5097-99, Task 2059," October 31, 2006.
3. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.
4. WCAP-16168-NP-A Revision 2, "Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval," June 2008.
5. NUREG-1874, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)," 3/1/07.
6. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004.
7. NRC Regulatory Guide 1.150, Revision 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," February 1983.

RR-3-43(I) Attachment 1: Additional Analysis for the TWCF Parameter

Purpose

The change in risk analysis for the Westinghouse pilot plant in WCAP-16168-NP-A, Revision 2, Beaver Valley Unit 1, was performed at 60 EFPY with the intention of bounding the embrittlement of all Westinghouse plants as determined using the through wall cracking frequency (TWCF) correlation in Reference 1. As shown in Table 1 of RR-3-43(I), the TWCF value calculated for Indian Point Unit 3 is not bounded by the Westinghouse pilot plant value for 60 EFPY. However, the TWCF calculated for the Westinghouse pilot plant at the EFPY for Condition B ("Ext-Bb") in Reference 2 is 9.33E-07 events per year. This value is bounding of the Indian Point Unit 3 TWCF value. Condition B is well beyond 60 EFPY so that the embrittlement levels will provide a TWCF that exceeds the Regulatory Guide 1.174 (Reference 3) risk limit of 1.0E-07 events per year. The purpose of the analysis documented in this attachment is to perform the change in risk calculation for the Westinghouse pilot plant at the EFPY for condition B. If the change in risk at this EFPY is acceptable, then the Westinghouse pilot plant bounds Indian Point Unit 3 for the TWCF parameter.

Method Discussion

The analysis performed in this attachment is consistent with that performed for the Westinghouse pilot plant in Reference 2. These calculations are performed at the EFPY for Condition B. All inputs are consistent with those used in Reference 2 with the exception that the fluence values in the FAVPFM input file correspond to the EFPY for Condition B rather than 60 EFPY. This input file is consistent with that used for Condition B in the NRC PTS Risk Re-Evaluation (Reference 1).

Results

Consistent with Reference 2, the FAVOR Code evaluation was run for each analysis for a number of vessel simulations required to obtain a stable solution. A summary of failure frequency results is included in Table 1 for the FAVPOST output for the 10 Year ISI only and 10 Year ISI Interval runs for Beaver Valley Unit 1.

Table 1: Failure Frequency Results (Events per Year)	
Case	Beaver Valley Unit 1 – Condition B
10 Year ISI Only (Mean Value)	5.08E-07
10 Year ISI Only (Standard Error)	8.23E-09
10 Year Interval (Mean Value)	5.09E-07
10 Year Interval (Standard Error)	9.69E-09

For delta risk calculations, the difference is typically taken between mean values. Consistent with Reference 2, an upper and lower bound were determined for the "10 Year ISI Interval" and "10 Year ISI Only" cases. The Upper Bound is calculated by adding two times the Standard Error to the "10 Year ISI Only" mean failure frequencies. The Lower Bound is calculated by subtracting two times the Standard Error from the "10 Year ISI Interval" mean failure frequencies. The change in failure frequency is conservatively calculated by taking the difference between these upper and lower bounds at the last vessel simulation. Table 2 displays the change in failure frequency results at the last vessel simulation. Figure 1 shows the change in failure frequencies calculated for different numbers of iterations or vessel simulations.

Table 2: Change in Failure Frequency Results – 10 Year ISI Interval and 10 Year ISI Only (Events per Year)	
Case	Beaver Valley Unit 1 – Condition B
Upper Bound	5.25E-07
Lower Bound	4.90E-07
Delta Risk (Mean Values)	-7.80E-10
Bounding Delta Risk	3.51E-08

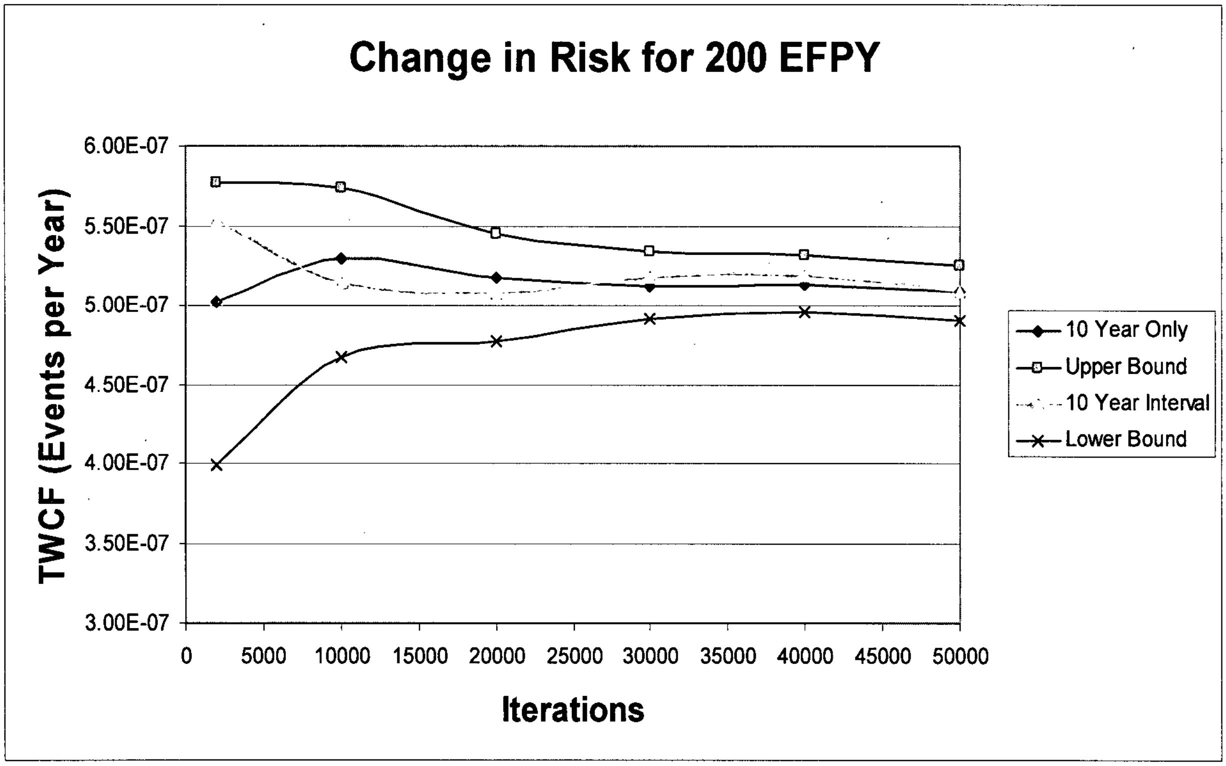

Figure 1: Comparison of Results for Beaver Valley Unit 1 10 Year ISI Interval and 10 Year ISI Only

Conclusions

In Regulatory Guide 1.174 (Reference 3), the acceptable change in risk was specified as 1.0E-7 events per year. As seen in Table 2, the bounding change in risk (delta risk) is below the acceptable change. Therefore, the conclusions of Reference 3 remain valid for the Westinghouse pilot plant (Beaver Valley Unit 1) at Condition B and for Indian Point Unit 3 at 60 EFPY.

References

1. NUREG-1874, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)," 3/1/07.
2. WCAP-16168-NP-A, Revision 2, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," June 2008.
3. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.