UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 3, 2009

Mr. Samuel L. Belcher
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT NO. 1 – REQUEST TO USE ALTERNATIVE FOR VOLUMETRIC EXAMINATION OF REACTOR PRESSURE VESSEL CIRCUMFERENTIAL SHELL WELDS FOR THE LICENSE RENEWAL PERIOD OF EXTENDED OPERATION (TAC NO. MD9704)

Dear Mr. Belcher:

By letter dated September 16, 2008, as supplemented by letter dated March 5, 2009, Nine Mile Point Nuclear Station, LLC (NMPNS, the licensee) submitted relief request (RR) 1ISI-001A, proposing an alternative to the requirements of Title 10 of the *Code of Federal Regulations*, Part 50, Section 55a (10 CFR 50.55a), concerning the requirements of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," for performing reactor pressure vessel (RPV) shell circumferential weld examinations for Nine Mile Point, Unit No. 1 (NMP1). The request would allow use of the proposed alternative for the license renewal period of extended operation (i.e., from August 23, 2009, to August 22, 2029). The fourth 10-year inservice inspection (ISI) interval will begin on August 23, 2009, concurrent with the NMP1 license renewal period of extended operation.

The Nuclear Regulatory Commission (NRC) staff has reviewed NMPNS's regulatory and technical analysis in support of 1ISI-001A. Based on the information provided by NMPNS, the NRC staff has concluded that the proposed alternative will provide an acceptable level of quality and safety for performing RPV shell circumferential weld examinations for NMP1. The licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the NMP1 license renewal period of extended operation.

The NRC staff's safety evaluation is enclosed. If you have any questions, please contact Rich Guzman at (301) 415-1030 or via email at Richard.Guzman@nrc.gov.

Sincerely,

Nancy L. Salgado

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosure:
As stated

cc w/encl: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE EXTENSION OF PERMANENT RELIEF FROM

INSERVICE INSPECTION REQUIREMENTS FOR THE VOLUMETRIC EXAMINATION OF

REACTOR PRESSURE VESSEL SHELL CIRCUMFERENTIAL WELDS FOR

THE LICENSE RENEWAL PERIOD OF EXTENDED OPERATION

NINE MILE POINT NUCLEAR STATION, LLC

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

By letter dated September 16, 2008 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML082671120), as supplemented by letter dated March 5, 2009 (ADAMS Accession No. ML090760978), Nine Mile Point Nuclear Station, LLC (NMPNS, the licensee) submitted relief request (RR) 1ISI-001A, proposing an alternative to the requirements of Title 10 of the *Code of Federal Regulations*, Part 50, Section 55a (10 CFR 50.55a), concerning the requirements of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," for performing reactor pressure vessel (RPV) shell circumferential weld examinations for Nine Mile Point, Unit No. 1 (NMP1). The request would allow use of the proposed alternative for the license renewal period of extended operation (i.e., from August 23, 2009, to August 22, 2029).

2.0 REGULATORY EVALUATION

The inservice inspection (ISI) of the ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda, as required by 10 CFR 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). Section 50.55a(a)(3) of 10 CFR Part 50 states, in part, that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, 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 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), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice 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 regulations require that inservice examination of components and system pressure tests conducted during the first 10-year ISI interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, which was 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.

Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Techniques for Determining Pressure Vessel Neutron Fluence," describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence with respect to the General Design Criteria (GDC) contained in Appendix A to 10 CFR 50. By considering the applicable GDC, the NRC staff establishes that the neutron fluence calculation adequately supports the requested inservice inspection (ISI) relief request.

In consideration of the guidance set forth in RG 1.190, GDC 14, 30, and 31 are applicable. GDC 14, "Reactor Coolant Pressure Boundary," requires the design, fabrication, erection, and testing of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires, among other things, that components comprising the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," pertains to the design of the reactor coolant pressure boundary, stating:

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a no brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

The ASME Code of record for the NMP1 fourth 10-year ISI interval program is the 2001 Edition through the 2003 Addenda of Section XI of the ASME Code. The fourth 10-year ISI interval will begin on August 23, 2009, concurrent with the NMP1 license renewal period of extended operation.

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 Boiling Water Reactor Vessel and Vessel Internals Project (BWRVIP), a technical committee of the BWR Owners Group, submitted the BWRVIP-05 report, "BWR Reactor Vessel Shell Weld Inspection Recommendations," 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 the 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 zero 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 NRC staff issued a safety evaluation (SE) on the BWRVIP-05 report[1]. 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 original 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[2] and 2.6-5[2] of the NRC staff's July 28, 1998, SE. 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, and Babcock and Wilcox. 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 NRC staff's SE identifies 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.

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1. The NRC staff has identified that in some instances, the staff SE report is referenced with a date of July 28, 1998, others with a date of July 30, 1998. For clarification purposes the staff notes that this SE report 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)."

2. Tables 2.6-4 and 2.6-5 are not included in this SE and can be found in the staff's SE report dated July 30, 1998.

2.1.2 Generic Letter 98-05

On November 10, 1998, the NRC staff issued Generic Letter (GL) 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report To Request Relief From Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," which 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 welds (ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, "Circumferential Shell Welds") by demonstrating that:

(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 NRC staff's July 28, 1998, SE report, and,

(2) Licensees have implemented operator training and established procedures that limit the frequency of LTOP events to the amount specified in the NRC staff's July 28, 1998, SE.

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

2.1.3 BWRVIP-74 Report

The NRC staff reviewed BWRVIP-74, "BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," dated September 21, 1999, to determine the applicability of this alternative for an extended period of operation (under license renewal). The NRC staff's evaluation of BWRVIP-74 was provided by SE dated October 18, 2001 (ADAMS Accession No. ML012920549), which concluded that Appendix E of the July 28, 1998, SE, conservatively evaluated BWR RPVs to 64 effective full-power years (EFPY), which is 10 EFPY greater than what is realistically expected for the end of an additional 20-year license renewal period. Therefore, the NRC staff's analysis provides a technical basis for relief from the current ISI requirements of the ASME Code Section XI for volumetric examination of the circumferential welds as they may apply for the license renewal period. The October 18, 2001, SE, further states that to obtain relief, each licensee will have to demonstrate that:

(1) At the end of the renewal period, the circumferential welds will satisfy the limiting conditional failure probabilities for circumferential welds in the Appendix E of the NRC staff's July 28, 1998, SE, and,

(2) That they have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the NRC staff's July 28, 1998, SE.

The July 28, 1998, SE, provides Table 2.6-5 which also includes the conditional failure probabilities identified in Appendix E for each vessel fabricator, along with the corresponding highest projected reference temperature. Therefore, this table provides the limiting case studies for plants requesting relief to the end of the extended period of operation. This relief does not apply to the axial welds, and therefore, the licensee is expected to perform the required ASME Code inspections of "essentially 100 percent" of all axial welds.

3.0 TECHNICAL EVALUATION

3.1 RR No. 1ISI-001A

3.1.1 ASME Code Component Identification

Pressure Retaining Reactor Pressure Vessel (RPV) Shell Circumferential Welds

Circumferential Shell Weld Numbers	Description	ASME Code Category	ASME Code Item Number
RVWD-100	Circumferential Shell Weld	B-A	B1.11
RVWD-101	Circumferential Shell Weld	B-A	B1.11
RVWD-137	Circumferential Shell Weld	B-A	B1.11
RVWD-138	Bottom Head-to-Shell Weld	B-A	B1.11

3.1.2 ASME Code Requirements

ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11 requires a volumetric examination of essentially 100 percent of the weld length of the RPV shell circumferential welds each inspection interval.

3.1.3 Licensee's Proposed Alternative Examination

In its letter dated September 16, 2008, the licensee stated it will implement the alternate provisions for the subject weld examinations in accordance with 10 CFR 50.55a(a)(3)(i), and consistent with NRC GL 98-05 and the NRC staff SE for BWRVIP-74-A as follows:

> The failure frequency for ASME Code Section XI, Table IWB-2500-1 Examination Category, B-A, Item No. B1.11, "Reactor Pressure Vessel Shell Circumferential Welds," is sufficiently low to justify their elimination from the in-service inspection (ISI) requirement of 10 CFR 50.55a(g) based on the NRC Safety Evaluation.
>
> The ISI examination requirements of the ASME Code, Section XI, Table IWB-2500-1 Examination Category B-A, Item No. B1.12, "Reactor Pressure Vessel Shell Longitudinal Welds," shall be performed, to the extent possible, and shall include inspection of the RPV Shell Circumferential Welds only at the intersection of these welds with the longitudinal welds, or approximately 2 to 3 percent of the RPV shell circumferential welds. The proposed alternative for volumetric examination of the RPV shell welds includes performing an examination, from the external outside diameter (OD) surface or where access is practical from the internal inside diameter (ID) surface of the RPV to the maximum extent possible. The examination of the remaining accessible portions of the RPV circumferential shell welds will be permanently deferred for the life of the original license and the license renewal period of extended operation.
>
> The procedures for these examinations shall be qualified such that flaws relevant to the RPV integrity can be reliably detected and sized, and the personnel implementing these procedures shall be qualified in the use of these procedures. Qualification and examination will be completed in accordance with the 2001 Edition through 2003 Addenda of ASME [Code,] Section XI, Appendix VIII as

modified by the Performance Demonstration Initiative (PDI) and 10 CFR 50.55(a), "Codes and Standards."

3.1.4 Licensee's Basis for Relief Request

In its letter dated September 16, 2008, the licensee stated that that permanent deferral of the examination of the RPV circumferential shell welds for the license renewal period of extended operation and the reduced examination coverage of the circumferential welds are justified and present an acceptable level of quality and safety to satisfy the requirements in accordance with 10 CFR 50.55a(a)(3)(i). The following summary items were provided by the licensee to support the licensee's basis for its relief request.

- The licensee provided a comparison of the NMP1 chemistry values and adjusted reference temperature (ART) against the NRC staff's July 28, 1998, SE, values for 32 EFPY.

- The NMP1 specific chemistry, and adjusted reference temperature (ART) were compared against the NRC's July 28, 1998, safety evaluation values for 32 EFPY. The NMP1 values were found to be bounded demonstrating that the NRC SE conclusions regarding failure probability have been satisfied.

- An NMP1 specific probabilistic fracture mechanics evaluation was performed to determine the probability of failure when subjected to an LTOP event during the license renewal period.

- The licensee confirmed that NMP1 has taken steps to reduce the potential for LTOP events through procedural controls and personnel training.

- An evaluation to identify the sources for increased pressure was also performed and found that the probability of a cold overpressure transient is considered to be less than or equal to that used in the NRC evaluation.

- The criterion in RG 1.174 regarding defense-in-depth, and safety margins are maintained and NRC safety goals are not exceeded.

The licensee provided its basis for the proposed RR 1ISI-001A by responding to the two criteria stated in GL 98-05 and the staff's SE for the review of BWRVIP-74:

Criterion (1) - Conditional Failure Probability

Demonstrate that at the expiration of the license (initial and renewed), the RPV shell circumferential welds will continue to satisfy the limiting conditional failure probability for RPV shell circumferential welds that is established in the July 28,1998 Safety Evaluation.

Licensee's Response

In order to demonstrate that the circumferential welds satisfy the July 28, 1998, NRC safety evaluation limiting conditional failure probabilities, a comparison of the chemistry values and the predicted fluence at the end of the original license period can be made. Note that the NMP1 current license period is equivalent to 28 EFPY. However, for the purpose of the inspection relief for the initial 40-year license, NMP1 used values for 32 EFPY to compare against the NRC 32 EFPY values presented in the SER. In addition, failure probabilities are also calculated for NMP1 at 46 EFPY, which corresponds to the end of the license renewal period of extended operation. For the license renewal period, it is more appropriate to compare the change in failure probabilities since the NRC analysis did not consider the effects of the license renewal period (added fluence and crack growth). In this evaluation, the change in risk is the governing factor determined by the difference between the probability of failure at the end of the license renewal period (46 EFPY) and the end of the original license period (28 EFPY). The NMP1 request for relief for the original license period is given in Reference 11, and the NRC's authorization is documented in Reference 13.

For the original license period, Table 1 illustrates that NMP1 has conservatism in comparison to NRC Final Evaluation of BWRVIP-05 Limiting Plant Specific Analysis (comparing 32 EFPY values). The chemistry factor, adjustment for reference temperature (ΔRT_{NDT}), and mean RT_{NDT}, are calculated consistent with the guidelines of NRC Regulatory Guide 1.99, Rev. 2 (Reference 16). The data presented for NMP1 in the BWRVIP response to the NRC Request for Additional Information (RAI) on BWRVIP-05 is also shown in Table 1. The maximum Cu percent and Ni percent variability from the most current data available is also bounded. The fluence values in Table 1 for 28 EFPY and 46 EFPY (from Reference 15) bound the highest fluence beltline circumferential weld, and were calculated using methods that are in accordance with Regulatory Guide 1.190 (Reference 17) and have been previously reviewed and approved by the NRC (References 5 and 6).

Table 1: Comparison of Input Parameters for NRC Staff Assessment and BWRVIP Methodology

Parameter Description	Nine Mile Point 1 (Circumferential Weld)			NRC Staff Assessment for 32 EFPY (Ref. 2) (Circumferential Weld)		Nine Mile Point 1 (Axial Weld)	
	Using BWRVIP Methodology			Safety Evaluation "VIP"	Safety Evaluation "CEOG"	Using BWRVIP Methodology	
	28 EFPY**	32 EFPY*	46 EFPY**	32 EFPY	32 EFPY	28 EFPY**	46 EFPY**
Fluence, n/cm^2	$1.6x10^{18}$	$2.21 x10^{18}$	$1.67x10^{18}$	$2x10^{18}$	$2x10^{18}$	$1.65x10^{18}$	$2.49 X10^{18}$
Initial RT_{NDT} °F	-50	-50	-50	0	0	-50	-50
Chemistry Factor	99.9	112	99.9	151.7	172.2	97.6	97.6
Cu%	0.214	0.22	0.214	0.13	0.183	0.214	0.214
Ni%	0.076	0.20	0.076	0.71	0.704	0.046	0.046
ΔRTNOT °F	44.7	66.5	52.7	86.4	98.1	51.2	60.8
Mean ART °F	-5.3	16.5	2.7	86.4	98.1	1.2	10.8

*From Reference 11. Note that the weld heat number 1248 chemistry used for the 32 EFPY calculations was revised for the 28 EFPY and 46 EFPY calculations based on the resolution of the NMP1 surveillance capsule weld identity as discussed in Reference 14. ** From Reference 7, 14, and 15.

As shown in Table 1, the impact of irradiation results in a lower plant specific mean RT_{NDT} for the NMP1 circumferential weld material as compared to that for any of the NRC's plant-specific analyses which were performed for the Combustion Engineering (CE) fabricated RPVs with the highest adjusted reference temperatures. Comparison of the NMP1 specific data and the data used in the NRC Final Safety Evaluation indicates that the combined effects of the Ni percent and Cu percent on the Chemistry Factor, which is by it self bounded by the NRC Independent Assessment, and the initial RT_{NDT}. Therefore, the limiting plant-specific conditional probability of failure P(FIE), determined by the NRC, bounds the NMP1 case through the projected end of the original license period.

Thus, for the original license period, the BWRVIP specific results relative to NMP1 as presented in BWRVIP-05 and subsequent RAI responses are consistent with those in the NRC Independent Assessment. Both analyses conclude that the failure probability associated with the circumferential welds is extremely small, and is orders of magnitude less than that for axial welds. Therefore, the NMP1 circumferential weld satisfies, at the end of the original license period, the limiting conditional failure probability for circumferential welds stated in the NRC's July 28, 1998, Safety Evaluation. Note that the discussion above is applicable for the original license period.

For the license renewal period, an NMP1 specific probabilistic fracture mechanics (PFM) evaluation was performed with the VIPER Program (Reference 8) using the data under the column "Using BWRVIP Methodology" in Table 1 for 28 EFPY and 46 EFPY (end of license renewal period). This evaluation was performed using the VIPER probabilistic fracture mechanics program that was developed as part of the BWRVIP-05 (Reference 1) effort. The same low temperature over pressure (LTOP) event parameters (Temperature = 88°F, Pressure = 1150 psi [pounds per square inch]) used in the BWRVIP-05 effort were used in this NMP1 specific calculation. Using the BWRVIP methodology the conditional probability of failure for the NMP1 circumferential weld was found to be less than $1x10^{-7}$ for 28 EFPY and 46 EFPY (no failures predicted in 10^7 trials for both 28 EFPY and 46 EFPY). The BWRVIP frequency of over-pressurization was determined to be $1\ x10^{-3}$/yr. This gives a total probability of failure for NMP1 of less than $2.5x10^{-12}$/yr for the circumferential welds for 28 EFPY (40 years) and 46 EFPY (60 years) of operation. Note that the failure probabilities are reported to be the same for 28 EFPY and 46 EFPY since there were no failures in 10^7 Monte Carlo trial simulations. The 46 EFPY includes higher fluence and considers crack growth for 18 EFPY beyond the original license period (28 EFPY).

For NMP1 axial welds with the data shown in Table 1 under the column "Using BWRVIP Methodology," the total probability is $<2.5x10^{-11}$/yr for both 28 EFPY and 46 EFPY, as no failures were predicted for either the original license period or the license renewal period.

The fact that no failures occurred through the initial license period and license renewal period shows that the reliability of the NMP1 RPV is extremely high. Most importantly, the results show that the increase in failure probability due to the license renewal period is essentially negligible. In addition, the reliability of the circumferential welds is likely much higher than calculated because of lower stress (axial stress is one half the hoop stress) and lower chemistry factors.

These calculations have been performed conservatively assuming a constant fluence at all weld locations equal to the peak fluence. For example, the maximum fluence anywhere in the beltline circumferential weld is assumed to exist throughout the circumferential weld. The peak fluence at any axial weld is assumed to exist at all axial weld locations. In reality, the fluence varies both circumferentially and axially. If analysis were performed considering these fluence variations, the resulting probability of failures would be lower than calculated using the peak fluence at all weld locations.

Thus, the BWRVIP-05 NMP1 specific results as determined using the BWRVIP-05 methodology and subsequent BWRVIP responses to NRC RAIs are consistent with those in the NRC Independent Assessment. Both analyses conclude that the failure probability associated with circumferential welds is extremely small. In addition, due to the NMP1 specific conditions, the failure probability for the axial welds is also extremely small. Most importantly, the increase in failure probability due to operation during the license renewal period is extremely small since no failures were predicted even through the license renewal period. Thus, it is concluded that the NMP1 circumferential weld satisfies, at the end of the license renewal period, the limiting conditional failure probability for circumferential welds in the NRC staffs July 28, 1998 safety evaluation.

Criterion (2) - Limiting the Frequency of Cold Over-pressure Events

Demonstrate licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC staff's July 28, 1998, safety evaluation.

Licensee's Response

The NRC indicated that the potential for, and consequences of, non-design basis events not discussed in the BWRVIP-05 report should be addressed. In particular, the NRC stated that non-design basis, low temperature over-pressure transients (LTOP) transients, should be considered. The NRC further went on to describe several types of events that could be precursors to an LTOP event. The BWRVIP provided a response to this issue concluding that Condensate and Control Rod Drive (CRD) pumps could cause such a condition leading to an LTOP event. This was summarized in the NRC Safety Evaluation for BWRVIP-05 (Reference 2).

NMP1 has in place procedures which monitor and control reactor pressure, temperature, and water inventory during all aspects of cold shutdown minimizing the likelihood of an RPV LTOP event. Additionally, these procedures are reinforced through the NMP1 reactor operator training program. The procedural controls and training provisions that will be used for the license renewal period of extended operation will be the same as those used for the original operating license period for NMP1 (see Reference 11 and 13).

The RPV leakage pressure test procedures used at NMP1 have sufficient procedural guidance to prevent LTOP events. The leakage test is performed at the conclusion of each refueling outage. These pressure tests are infrequentlyperformed, complex tasks, and the test procedures are controlled as Special Plant Evolutions. As such, a requirement is included in the procedures for an extensive pre-job briefing to be conducted with all essential personnel including Operations management. The briefing details the anticipated testing evolution with special emphasis on conservative decision making, plant safety awareness, lessons learned from similar in-house or industry operating experiences, the importance of open communications and finally the process in which the test would be aborted if plant systems responded in an adverse manner. Vessel pressure and temperature are required to be monitored throughout the tests to ensure compliance with the plant Technical Specification pressure-temperature curve. Also, the procedures require the designation of a "Principal Test Engineer" for the duration of the test that is a single point of accountability, responsible for the coordination of testing from initiation to closure, and maintaining operations and plant management cognizant of the test status.

With regard to inadvertent system injection resulting in an LTOP condition, the NMP1 high pressure make-up system, (i.e., the High Pressure Coolant Injection (HPCI)) and the normal Feedwater system are interconnected. The HPCI system is a mode of operation of the Condensate and Feedwater systems rather than an independent, stand alone system. The HPCI system utilizes two condensate pumps, two feedwater booster pumps, two motor-driven feedwater pumps, and an integrated control system. As such, the HPCI system contains only instrumentation and control components as its own dedicated equipment. HPCI initiation is prompted by the Reactor Protection System under the following conditions: (1) a turbine trip, or (2) low reactor water level. During shutdown of the unit, the associated booster and feedwater pumps in the system are secured in accordance with operating procedures. Equipment malfunction or inappropriate operational action would be necessary to cause inadvertent system operation.

During normal cold shutdown conditions, with the RPV head installed, RPV level and pressure are controlled with the CRD System, Condensate Feedwater System, and Reactor Water Cleanup (RWCU) systems using a "feed and bleed" process. The RPV is not taken solid during these times, and plant procedures require opening of the head vent valves after the reactor has been depressurized to approximately 15 psig [pounds per-square inch gauge.]

The Liquid Poison System is another high pressure water source to the RPV; however, there are no means of automatic system activation. System injection requires an operator to manually reposition a key-locked control switch to start the system from the Control Room. The system may also be operated from a remote local test station. The only injection path to the RPV is through two explosive actuated injection valves that are interlocked with the key-locked switch in the Control Room. The injection rate for each pump is approximately 30 gpm, which would give the operator sufficient time to control reactor pressure. Local testing of the pumps uses demineralized water from a test tank and is a closed test loop.

Procedural controls are in place to respond to an unexplained rise in reactor pressure which could result from a spurious activation of an injection source.

Actions specified include determination and isolation of the pressure source, verification of reactor head vents and/or MSIVs [Main Steam Isolation Valve] open and, as necessary, relieving reactor pressure using available plant equipment (e.g., electromagnetic relief valves, reactor water cleanup system and reactor bottom drain).

During normal cold shutdown conditions, reactor water level and temperature are maintained within established ranges in accordance with operating procedures. Procedures governing the conduct of operations require that the Control Room operators frequently monitor for indications and alarms to detect problems and abnormalities as early as possible. An Operations procedure also requires-that the control room supervisor be notified immediately of any change or abnormality in plant indications and controls. Furthermore, reactor water level and temperature operating bands and changes thereto are established under the direction of the Shift Manager. Therefore, any deviations in reactor water level or temperature from a specified band will be identified and corrected. Finally, plant conditions and on-going activities are discussed during each shift turnover. This ensures that on-coming operators are cognizant of activities that could adversely affect reactor level, pressure, or temperature.

Plant specific procedures have been developed to provide operator guidance regarding compliance with the plant Technical Specifications and RPV pressure-temperature curve limits. Additionally, operators receive training on RPV brittle fracture and the relationship of these pressure-temperature curve limits.

During plant outages, NMP1 work control processes ensure that the outage schedule and changes to the schedule receive a thorough shutdown risk assessment review to ensure defense-in-depth is maintained. Work is coordinated through the Work Execution Center which provides an additional level of Operations oversight. In the Control Room, the Shift Manager is required, by procedure, to maintain cognizance of any activity that could potentially affect reactor safety during refueling outages. Expected plant responses and contingency actions to address unexpected conditions that may be encountered are required to be evaluated as stated in the administrative controls for risk management and management of outages.

As discussed above, NMP1 has implemented procedural controls and training to minimize the probability of an LTOP event. Accordingly, the above information and the supporting technical documentation contained in the BWRVIP-05 report and NRC Safety Evaluation provide a basis for excluding RPV circumferential welds from the augmented examination requirements of 10 CFR 50.55a(g) and ASME [Code,] Section XI.

The licensee also provided the following additional information in its letter dated March 5, 2009.

<u>NRC Question RAI-1</u>

It is stated in the NMPNS submittal that the fluence projection covering the end of the facility operating license is based on the accrual of 46 EFPY of exposure. However, certain passages of the NMPNS license renewal application, dated July 14, 2005, presented limiting fluence projections based on accrual of 54 EFPY of exposure. Please explain this apparent inconsistency.

Licensee's Response

In Section 4.2.1 of the NMPNS license renewal application (Reference 1), a reactor vessel irradiation value of 54 Effective Full Power Years (EFPY) was used in the assessment of NMP1 RPV upper-shelf energy. This value was based on 60 years of plant operation with an average capacity factor of 90 percent, and was consistent with the value used in the generic evaluations contained in the Boiling Water Reactor Vessel and Internals Project technical report BWRVIP-74-A (Reference 2).

NMP1 has not operated at an average capacity factor of 90 percent since the time of initial licensing. The NMP1 evaluations contained in license renewal application Section 4.2.2, "Pressure-Temperature (P-T) Limits," were based on a reactor vessel irradiation of 46 EFPY. This value was determined by adding irradiation corresponding to an assumed average capacity factor of 90 percent during the 20-year period of extended operation to the 28 EFPY of exposure that was projected for the end of the original 40-year license term. An exposure of 46 EFPY continues to be projected for the end of 60 years of plant operation and was therefore used as the basis for the analyses performed to support 10 CFR 50.55a request number 1-ISI-001A.

NRC Question RAI-2

The enclosure to the NMPNS submittal states, "The fluence values in Table I for 28 EFPY and 46 EFPY (from Reference 15) bound the highest fluence beltline circumferential weld... " Please clarify this statement. How was it determined that the listed fluence values bound the highest fluence beltline circumferential weld? What is the location that corresponds to the listed fluence value?

Licensee's Response

The fluence values listed in Table 1 of 10 CFR 50.55a request number 1ISI-001A for 28 EFPY and 46 EFPY of exposure were obtained from the calculation documented in Reference 15 of the request. The fluence calculations were performed using methods that are in accordance with Regulatory Guide 1.190, and have been previously reviewed and approved by the NRC (as documented in References 5 and 6 of request number 1ISI-001 A). The Table 1 fluence values bound the peak values calculated in Reference 15 at any point on the circumferential weld between the lower and lower intermediate courses. This weld is located within the beltline region; 29.8 in. above the bottom of active fuel (weld no. RVWD-137 at elevation 239 ft-0in. shown on Figure 1 of request number 1ISI-001A). The maximum calculated fluence for weld no. RVWD-137 occurs at an azimuthal angle of 17 degrees with octant symmetry. As discussed on page ISI 001A-5 of the request, the failure probability calculations were performed conservatively assuming that the maximum fluence anywhere in the beltline circumferential weld exists throughout the circumferential weld. In reality, the fluence varies circumferentially. If analyses were performed considering these fluence variations, the resulting probability of failures would be lower than calculated using the peak fluence at all weld locations.

3.2 NRC Staff's Evaluation

ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11 requires a volumetric examination of essentially 100 percent of the weld length of the RPV shell circumferential welds each inspection interval. The licensee requested permanent relief from the requirements of the ASME Code through the end of the current licensed period of operation and including an operating license extension of 20 years for NMP1.

As described previously, GL 98-05 and the NRC staff's October 18, 2001, SE for BWRVIP-74 provide two criteria that BWR licensees requesting 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) must satisfy. These criteria are intended to demonstrate that the conditions at the applicant's plant are bounded by those in the NRC staff's SE. The licensee will still need to perform the required inspections of "essentially 100 percent" of all axial welds through the end of the license renewal extended power of operation which expires on August 22, 2029.

3.2.1 Circumferential Shell Weld Conditional Failure Probability

The NRC staff provides guidance for calculating neutron fluence in RG 1.190. The NRC staff evaluated the licensee's conformance with Criterion 1 of GL 98-05 using the guidance provided in RG 1.190.

Regarding the fluence calculations, the licensee stated that "the fluence values for 28 EFPY and 46 EFPY bound the highest fluence beltline circumferential weld, and were calculated using methods that are in accordance with Regulatory Guide 1.190 and have been previously reviewed and approved by the NRC." In so stating, the licensee refers to an NRC-issued SE dated October 27, 2003 (ADAMS Accession No. ML032760696), in which Section 3.3 describes in detail the neutron transport calculations and their acceptability based on adherence to the guidance contained in RG 1.190. This review is described in sufficient detail in the NRC staff's October 27, 2003, SE, and is not repeated herewith.

The NRC staff requested additional information from the licensee to confirm the applicability of the fluence calculations to the requested relief extension. Specifically, the NRC staff requested that the licensee (1) confirm the applicability of a 46 EFPY fluence calculation to the end of extended life, which is typically 54 EFPY based on a 90-percent capacity factor, and (2) clarify the statement that the fluence projections bound the highest fluence beltline circumferential weld.

3.2.2 Evaluation of RAI Question 1 – Applicability of 46 EFPY Fluence Calculations

In its supplemental letter dated March 5, 2009, the licensee stated that a reactor vessel irradiation value of 54 EFPY had previously been submitted to the NRC to support the NMP1 license renewal. However, the licensee also stated that NMP1 has not operated at a 90 percent capacity factor, which formed the basis for a 54 EFPY fluence projection.

The licensee clarified further that an exposure of 28 EFPY was predicted for the end of the initial 40-year operating period, and that projecting a 90 percent capacity factor during the 20-year period of extended operation would result in 46 EFPY of neutron flux accrual. Because the licensee clarified that the operating history at NMP1 supports a 46 EFPY projection of exposure at the end of the extended operating life, the NRC staff concludes that 46 EFPY fluence projections reasonably bound the period of extended operation such that these fluence values are acceptable.

3.2.3 Evaluation of RAI Question 2 - Clarification of Highest Beltline Weld Statement

In its supplemental letter dated March 5, 2009, the licensee stated that the fluence values supporting the relief request bound explicitly calculated peak fluence values obtained using the neutron transport method described in the NRC staff's October 27, 2003, SE. The licensee clarified further that the failure probability calculations were performed conservatively assuming that the maximum fluence anywhere in the beltline circumferential weld exists throughout the circumferential weld. Since this assumption increases the failure probability, it is conservative, and acceptable to the NRC staff on that basis.

3.2.4 Minimizing the Possibility of Low Temperature Overpressurization

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

The licensee examined a number of conditions that may be precursors to cold overpressurization. Although no cold overpressurization events occurred in this country, the NRC staff has identified cold overpressurizations in other countries. Those events resulted from operator errors and/or lack of adequate procedures. The conditions examined in the submittal are all the circumstances that could be reasonably construed as overpressurization precursors.

- High Pressure Core Spray Injection (HPCSI) HPCSI is used to flush injection piping after the reactor has been shutdown during RPV cooldown. The procedure for this evolution has multiple contingencies to prevent overpressurization. Multiple operator errors or equipment malfunction would have to occur before overpressurization could occur. Operator training and good maintenance practices would make the event extremely unlikely.

- RPV level and Control Rod Drive (CRD) system operation During normal cold shutdown conditions, with the RPV head installed, RPV level and pressure are controlled with the CRD System, Condensate Feedwater System, and Reactor Water Cleanup (RWCU) systems using a "feed and bleed" process. The RPV is not taken solid during these times, and plant procedures require opening of the head vent valves when the coolant temperature is less than 212°F and after the reactor has been depressurized to approximately 15 psig. Under these conditions the CRD system injects at a rate of about

65[3] gallon-per-minute (gpm). Therefore, cold overpresurization due to inadvertent CRD system operation is highly unlikely. At this low rate, the operator has sufficient time to respond to any level and pressure changes. Therefore, cold overpresurization due to inadvertent CRD system operation is highly unlikely.

- The Liquid Poison System (LPS) is another high pressure injection source to the RPV. However, the LPS is normally shutdown and there are no automatic starts. LPS initiation requires control room or local manual action. In addition, the maximum injection rate is only 30 gpm thus affording the operator adequate time to identify and stop any change in vessel water level caused by inadvertent initiation of the LPS. Therefore, cold overpressurization due to LPS initiation is highly unlikely.

- RPV pressure testing is an infrequent evolution and it is performed under strict management supervision so as not to violate facility pressure/temperature limits. The licensee performs the leakage tests at the end of each refueling outage and is controlled as special plant evolutions. For example, extensive pre-job briefings are conducted for all essential personnel including Operations Management and includes the anticipated testing evolution with special emphasis on conservative decision making, plant safety awareness, lessons learned from similar in-house or industry operating experiences, the importance of open communications, and finally, the process in which the test would be aborted if plant systems responded in an adverse manner. The RPV pressure and temperature is monitored by the licensee during these tests to ensure they are in compliance with the plant Technical Specification pressure-temperature curves. Furthermore, a principal test engineer is appointed to ensure there is only one point of contact throughout the test for accountability, and who is responsible for the coordination of the testing from initiation to closure. The principal test engineer is also responsible for maintaining operations and informing plant management of the test status. These practices along with operator training minimize the possibility of vessel cold overpressurization due to pressure vessel testing.

The licensee's evaluation of potential injection sources that would cause an LTOP event is largely consistent with the industry response to the NRC staff's evaluation of the BWRVIP-05 report. Most high pressure injection sources that could cause LTOP events are prevented by interlocks, plant conditions, and/or administrative controls. In its final SE, regarding the BWRVIP-05 report, the NRC staff notes that the CRD system could cause conditions which could lead to LTOP events. However, as noted above, the injection rate of the CRD system is low and it will allow the operator sufficient time to react to unanticipated level changes which reduces the possibility of an event that would result in a violation of the pressure/temperature limits.

3.2.5 Operator Training and Operating Procedures

The licensee's submittal states that the procedural controls and training provisions that will be used for the license renewal period of extended operation will be the same as those used for the original operating license period for NMP1. The licensee's operator training is described below and is consistent with the description provided in the licensee's letter for NMP2 dated April 20,

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3 The 65 gpm for the CRD flow rate was obtained from the licensee's letter dated November 5, 1998 for a Licensee Event Report No, 98-18, Revision 00 for NMP1.

2007 (ADAMS ML0712102451), and the licensee's letter for NMP1 dated December 10, 1998 (Reference 11 in NMPNS's submittal dated September 16, 2008).

- During normal cold shutdown conditions, vessel water level, pressure, and temperature are maintained within predetermined bands in accordance with a procedure. The procedure requires that operators frequently monitor the parameters to detect abnormalities as early as possible and the shift supervisor be notified should such an abnormality appear. In addition, changes that may affect water level, temperature, or pressure can only be performed under the supervision of the shift supervisor. Finally, activities that may affect critical parameters are discussed in shift turnovers.

- Operators are trained in methods of controlling water level, pressure, and temperature if they are found outside specified limits. Also operators are trained in brittle fracture and maintaining the plant within the pressure temperature technical specification limits.

- During plant outages, the work control processes ensure that the outage schedule and its changes are reviewed to ensure that defense-in-depth is maintained. The outage schedule is under senior reactor operator supervision to avoid conditions that could adversely affect water level, temperature, and pressure.

- During plant outages work is coordinated through the work control center that provides an additional level of operational oversight. Pre-job briefings are contacted for complex work activities such as vessel pressurization that have the potential of affecting vessel parameters. Such briefings are attended by all those involved in the evolutions. In addition, expected plant responses and contingency plants are addressed in the briefing discussions.

Operator training covers, among other things, basic theory and application of brittle fracture, vessel thermal stresses, operational transient procedures, including high water level, and technical specification limitations (pressure/temperature limit curves) for heatup and cooldown. These scenarios are reinforced in simulator exercises and the required operational skills are developed. Such training also reinforces managements' expectations for procedural compliance by the operators.

4.0 CONCLUSION

Based on its review of the licensee's fluence calculations, the NRC staff concludes the following:

1. The NRC staff has previously reviewed the fluence calculations used to support the present relief request and concluded that they were performed in adherence to RG 1.190, and were acceptable on that basis. This finding remains unchanged for this relief request.

2. The licensee's fluence calculations include sufficient exposure to encompass the end of the plant's 60-year operating life.

3. The weld failure probabilities were estimated using a limiting fluence value.

In consideration of these three items, the NRC staff concludes that the licensee's fluence calculations are acceptable to support the requested ISI relief.

In addition, the NRC staff has concluded that the licensee has demonstrated conformance with the applicable safety evaluation criteria in NRC GL 98-05 and in the NRC staff's evaluation of the BWRVIP-05 and BWRVIP-74 reports. The NRC staff has also concluded that the licensee has acceptably demonstrated that the conditional probability of failure values for the NMP 1 circumferential welds are sufficiently low enough to justify elimination of the volumetric examinations that are required by the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11.

Based on the above analysis, the NRC staff concludes that the licensee's alternative will provide an acceptable level of quality and safety for performing RPV shell circumferential weld examinations for NMP1. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the NMP1 license renewal period of extended operation (i.e., August 23, 2009, to August 22, 2029). Furthermore, based on the considerations discussed above, the NRC staff concludes that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation of NMP1 in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this alternative will not be inimical to the common defense and security or the health and safety of the public.

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

Principal Contributors: T. K. McLellan
B. Parks

Date: August 3, 2009

August 3, 2009

Mr. Samuel L. Belcher
Vice President Nine Mile Point
Nine Mile Point Nuclear Station, LLC
P.O. Box 63
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT NO. 1 – REQUEST TO USE ALTERNATIVE FOR VOLUMETRIC EXAMINATION OF REACTOR PRESSURE VESSEL CIRCUMFERENTIAL SHELL WELDS FOR THE LICENSE RENEWAL PERIOD OF EXTENDED OPERATION (TAC NO. MD9704)

Dear Mr. Belcher:

By letter dated September 16, 2008, as supplemented by letter dated March 5, 2009, Nine Mile Point Nuclear Station, LLC (NMPNS, the licensee) submitted relief request (RR) 1ISI-001A, proposing an alternative to the requirements of Title 10 of the *Code of Federal Regulations*, Part 50, Section 55a (10 CFR 50.55a), concerning the requirements of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," for performing reactor pressure vessel (RPV) shell circumferential weld examinations for Nine Mile Point, Unit No. 1 (NMP1). The request would allow use of the proposed alternative for the license renewal period of extended operation (i.e., from August 23, 2009, to August 22, 2029). The fourth 10-year inservice inspection (ISI) interval will begin on August 23, 2009, concurrent with the NMP1 license renewal period of extended operation.

The Nuclear Regulatory Commission (NRC) staff has reviewed NMPNS's regulatory and technical analysis in support of 1ISI-001A. Based on the information provided by NMPNS, the NRC staff has concluded that the proposed alternative will provide an acceptable level of quality and safety for performing RPV shell circumferential weld examinations for NMP1. The licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the NMP1 license renewal period of extended operation.

The NRC staff's safety evaluation is enclosed. If you have any questions, please contact Rich Guzman at (301) 415-1030 or via email at Richard.Guzman@nrc.gov.

Sincerely,

/RA/
Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-220

Enclosure:
As stated

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